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ABSTRACT

This technical specialty course is one of thirty-five courses designed for use by two-year postsecondary institutions in five nuclear technician curriculum areas: (1) radiation protection technician, (2) nuclear instrumentation and control technician, (3) nuclear materials processing technician, (4) nuclear quality-assurance/quality-control technician, and (5) nuclear reactor (plant) operator trainee. This course, Radiation Shielding, contains six modules: General Effect of Radiation Attenuation, Beta and Alpha Particle Attenuation, Neutron Attenuation, Gamma Attenuation, Background Radiation Reduction Techniques, and Backscattering and Wall Scattering of Radiation. Each module follows a typical format including the following sections: introduction, module prerequisites, objectives, subject matter, exercises, materials, procedures, data tables, reference materials, and tests. (A master index and suggested program planning guides are also available--see CE 026 856 and CE 026 639-643.) (LRA)

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ED192101

Nuclear Technology Series

COURSE 19

RADIATION SHIELDING

CENTER FOR OCCUPATIONAL RESEARCH
AND DEVELOPMENT, INC.

4800 LAKEWOOD DRIVE
WACO, TEXAS 76710

CE 026 659

U S DEPARTMENT OF HEALTH,
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NATIONAL INSTITUTE OF
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JUL 1980

PREFACE

ABOUT NUCLEAR TECHNOLOGY MODULES

The modules – developed for use in two-year postsecondary technical institutions to prepare technicians for employment – are useful in industry for updating employees in company-sponsored training, and as laboratory texts in engineering schools. The principles, techniques and skills taught in the modules, based on tasks that nuclear technicians perform, were obtained from a nationwide advisory committee of employers of nuclear technicians. Each module was written by a working technical expert/experienced writer team and tested with students.

A module contains the following elements:

Introduction, which identifies the topic and often includes a rationale for studying the material.
Module Prerequisites, which identify the material a student should be familiar with before studying the module.
Objectives, which clearly identify what the student is expected to know for satisfactory module completion. The objectives, stated in terms of action-oriented behaviors, include such action words as operate, measure, calculate, identify and define, rather than words with many interpretations, such as know, understand, learn and appreciate.
Subject Matter, which presents the background theory and techniques supportive to the objectives of the module. Subject matter is written with the technical student in mind.
Exercises, which provide practical problems to which the student can apply this new knowledge.
Materials, which identify the equipment required to complete the laboratory procedure.
Procedures, which is the experiment section, or "hands-on" portion of the module (including step-by-step instruction) designed to reinforce student learning.
Data Tables, which are included in most modules for the first year (or basic) courses to help the student learn how to collect and organize data.
Reference Materials, which are included as suggestions for supplementary reading/viewing for the student.
Test, which measures the student's achievement of the prestated objectives.

A helpful Index has been added at the end of each course.

The materials contained herein were developed pursuant to a contract with the Office of Education, U.S. Department of Health, Education and Welfare: Contract No. 300780582. Contractors undertaking such projects under Government sponsorship are encouraged to express freely their professional judgment in the conduct of the project. Points of view or opinions stated do not, therefore, necessarily represent official Office of Education position or policy.

The Nuclear Technology Series consists of five curricula:

Nuclear Reactor (Plant) Operator Trainee.
Radiation Protection Technician.
Nuclear Materials Processing Technician.
Nuclear Quality-Assurance/Quality-Control Technician.
Nuclear Instrumentation and Control Technician.

The completed Nuclear Technology Series consists of the following 35 technical specialty courses:

Radiation Physics.
Radiation Protection I.
Principles of Process Instrumentation.
Heat Transfer and Fluid Flow.
Introduction to Nuclear Systems.
Instrumentation and Control of Reactors and Plant Systems.
Reactor Operations.
Reactor Safety.
Reactor Auxiliary Systems.
Power Plant Systems.
Radiation Detection and Measurement.
Reactor Physics.
Power Plant Chemistry.
Introduction to Quality Assurance/ Quality Control.
Metallurgy and Metals Properties.
Mechanical Component Characteristics and Specifications.
Radiation Protection II.
Radiological Emergencies.
Radiation Shielding.
Radiation Monitoring Techniques (Radiochemical).
Radioactive Materials Disposal and Management.
Advanced Radionuclide Analysis.
Nuclear Chemical Processes.
Nuclear Systems and Safety.
Radioactive Material Handling Techniques.
Nondestructive Examination (NDE) Techniques I.
Metrology.
Welding Inspection.
Civil/Structural Inspection.
Mechanical Inspection.
Quality-Assurance Practices.
Nondestructive Examination (NDE) Techniques II.
Control Systems I.
Control Systems II.
Power Plant Electrical Systems and Components.

Availability: Instructional modules are available individually or by courses from:

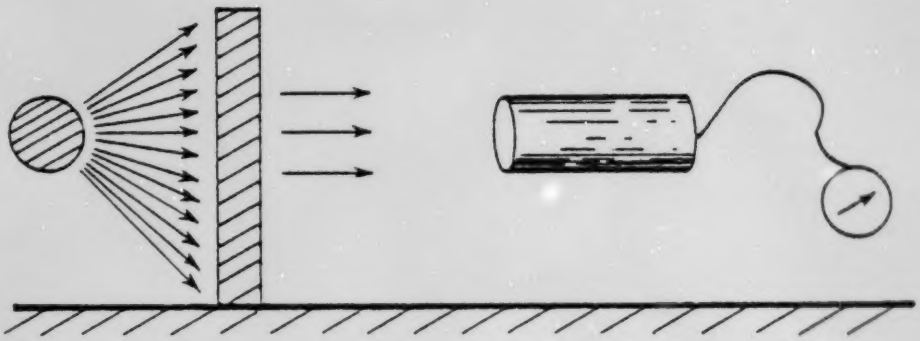
Center for Occupational Research and Development
4800 Lakewood Drive, Suite 5
Waco, Texas 76710
Telephone (817) 772-8756

Printed:

NUCLEAR TECHNOLOGY

MODULE 19 - 1

GENERAL EFFECT OF RADIATION ATTENUATION



RADIATION SHIELDING

COURSE 19

ORD CENTER FOR OCCUPATIONAL RESEARCH AND DEVELOPMENT

COURSE MODULES

COURSE 19 RADIATION SHIELDING

MODULE 19-1 General Effect of Radiation Attenuation
MODULE 19-2 Beta and Alpha Particle Attenuation
MODULE 19-3 Neutron Attenuation
MODULE 19-4 Gamma Attenuation
MODULE 19-5 Background Radiation Reduction Techniques
MODULE 19-6 Backscattering and Wall Scattering of Radiation

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INTRODUCTION

There are three general methods for the protection of persons or objects from the unwanted effects of ionizing radiation:

- Reduction of the time spent near a radiation source.
- Reduction of exposure to radiation by increasing the distance from a radiation source.
- Use of a radiation-absorbing material that attenuates the intensity of radiation.

This course explores the last of these methods – radiation shielding. The principles of shielding persons and objects from particulate, electromagnetic, and mixed radiation are discussed. The course describes the physical characteristics of particulate and electromagnetic radiation, and it examines the interaction of these types of radiation with absorbing materials. Methods employed by technicians for the determination of the necessary amounts of shielding also are presented.

This first module discusses the general effects of radiation attenuation and serves as an introduction to the other modules in the course. Some new concepts are introduced, and definitions of terms necessary for the discussion of shielding procedures are provided.

MODULE PREREQUISITES

The student should have completed a course in basic algebra; Course 1, "Radiation Physics"; Course 2, "Radiation Protection I"; and Course 11, "Radiation Detection Systems."

OBJECTIVES

Upon completion of this module, the student should be able to:

1. Cite the five factors that influence the attenuation of radiation in matter.
2. Define the following terms:
   - Linear absorption coefficient.
   - Mass absorption coefficient.
   - Half-value layer.
   - Tenth-value layer.
   - Cross section.
   - Barn.
   - Specific ionization.
   - Linear energy transfer.
   - Quality factor.
3. Given initial exposure rate, initial distance, and desired exposure rate, calculate the distance at which exposure rate will reach desired limits.
4. Given the type and intensity of radiation and a table of half-value layers, calculate the amount of lead shielding required to reduce the exposure rate to a specified amount.
5. Describe the relationship of specific ionization to mass, to energy, and to linear energy transfer.
6. Describe the relationship of linear energy transfer to biological effectiveness and to quality factor.
7. Given the type of source, exposure rate, and a table of quality factors, calculate total dose equivalent.

RADIATION ATTENUATION IN MATTER

"Attenuation" is the process by which a beam of radiation is reduced in intensity as it passes through a material. The Radiological Health Handbook defines attenuation as "the combination of absorption and scattering processes that leads to a decrease in the flux density of a beam when projected through matter." "Flux density" is the number of particles that pass through a unit of cross-sectional area per unit of time.

The degree and type of radiation attenuation is a function of the following factors:

- Type of radiation.
- Energy of the radiation.
- Physical properties of the absorbing medium.

TYPE OF RADIATION

Particulate radiation (alpha, beta, and neutron) has a definite range in matter. Alpha particles lose energy very rapidly in any medium through which they pass; for example, a 4-MeV alpha particle has a range in biological tissue of only a few hundredths of a millimeter and can travel only 3 centimeters in air. Beta particles have a range greater than that of alpha particles and can travel several feet in air. Neutrons can travel much farther than alpha or beta particles before losing all of their energy. Eventually, all of these particles are completely absorbed by the medium through which they travel.

Charged particles lose most of their energy during the course of a large number of collisions with atoms. The

larger an individual particle is the greater is the chance that it will collide with nuclei or orbital electrons. Alpha particles have a relatively great mass, and their path through matter is, essentially, a straight line; the path of the less massive beta particle is more erratic.

Alpha and beta particles carry an electrical charge, and electrical and magnetic forces affect their motion. These forces are so strong that charged particles penetrate matter very poorly. Neutrons have no charge and undergo different interactions. Their indirect ability to ionize comes about by collision with other atomic nuclei, which produce the ionization. Since the atoms in molecules are far apart in relation to neutron diameter, neutrons have infrequent collisions and, therefore, travel much farther than alpha and beta particles before losing all of their energy.

Some types of electromagnetic radiation (gamma and X-ray), unlike particulate radiation, are not absorbed completely by shielding. This type of electromagnetic radiation can penetrate extremely dense material, and it presents the greatest external radiation hazard. The intensity of these rays is reduced by shielding, but it is not attenuated entirely.

Electromagnetic radiation undergoes exponential attenuation as described by Equation 19-1-1:

$$I = I_0 e^{-\mu t} \qquad \text{Equation 19-1-1 (Equation = Eq.)}$$

where

I = intensity of radiation transmitted through a medium of thickness t.

I_0 = initial intensity of radiation (before absorption).
e = base of natural logarithms.
μ = attenuation coefficient in cm^{-1}.
t = thickness of medium in cm.

Figure 19-1-1 depicts the attenuation of electromagnetic radiation.

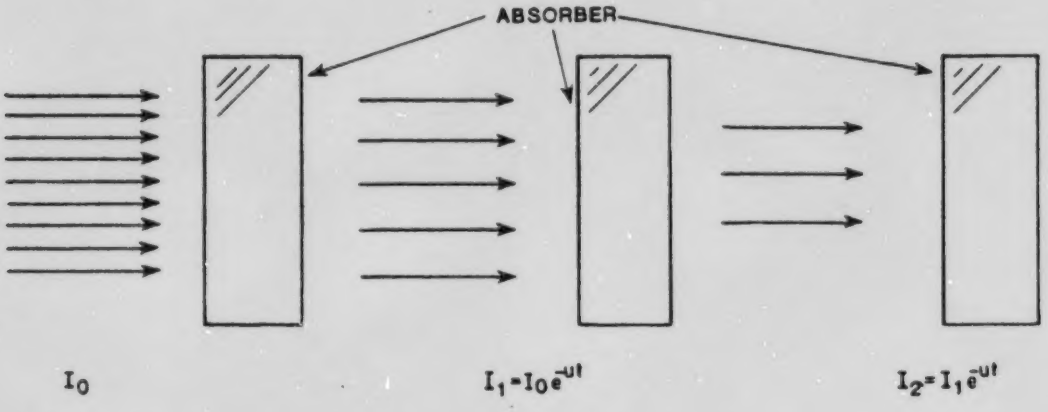


Figure 19-1-1. Attenuation of electromagnetic radiation.

The attenuation coefficient (μ) is the fractional decrease in the intensity of the beam per unit of thickness. Since a beam of electromagnetic radiation loses energy by a variety of processes (photoelectric effect, Compton scattering, pair production), the total attenuation coefficient is the sum of the individual attenuation coefficients of these processes.

If the units of the attenuation coefficient are cm^{-1}, the thickness of the medium is expressed in centimeters, and the attenuation coefficient is called the "linear" or "attenuation" coefficient. If the thickness is given in units of g/cm^2, the absorption coefficient is called the

"mass absorption coefficient." The relationship between linear and mass attenuation coefficients can be expressed by Equation 19-1-2:

$$\mu\ (cm^{-1}) = \mu_m\ (\frac{cm^2}{g}) \times \rho\ (\frac{g}{cm^3}) \qquad \text{Eq. 19-1-2}$$

This equation also can be expressed as Equation 19-1-3:

$$\mu_m = \frac{\mu}{\rho} \qquad \text{Eq. 19-1-3}$$

where in both cases,

μ = linear absorption coefficient.

ρ = density of the absorber.

μ_m = mass absorption coefficient.

Tables of linear and mass attenuation coefficients are given for various absorbing materials as a function of the energy of the radiation. Figure 19-1-2 and Table 19-1-1, from the Radiological Health Handbook, are typical.

Figure 19-1-2 illustrates mass attenuation coefficients for lead. Attenuation coefficients due to Compton scattering, the photoelectric effect, and pair production are shown; and a total attenuation coefficient curve is provided.

Table 19-1-1 depicts mass attenuation coefficients for various materials as a function of the energy of the radiation.

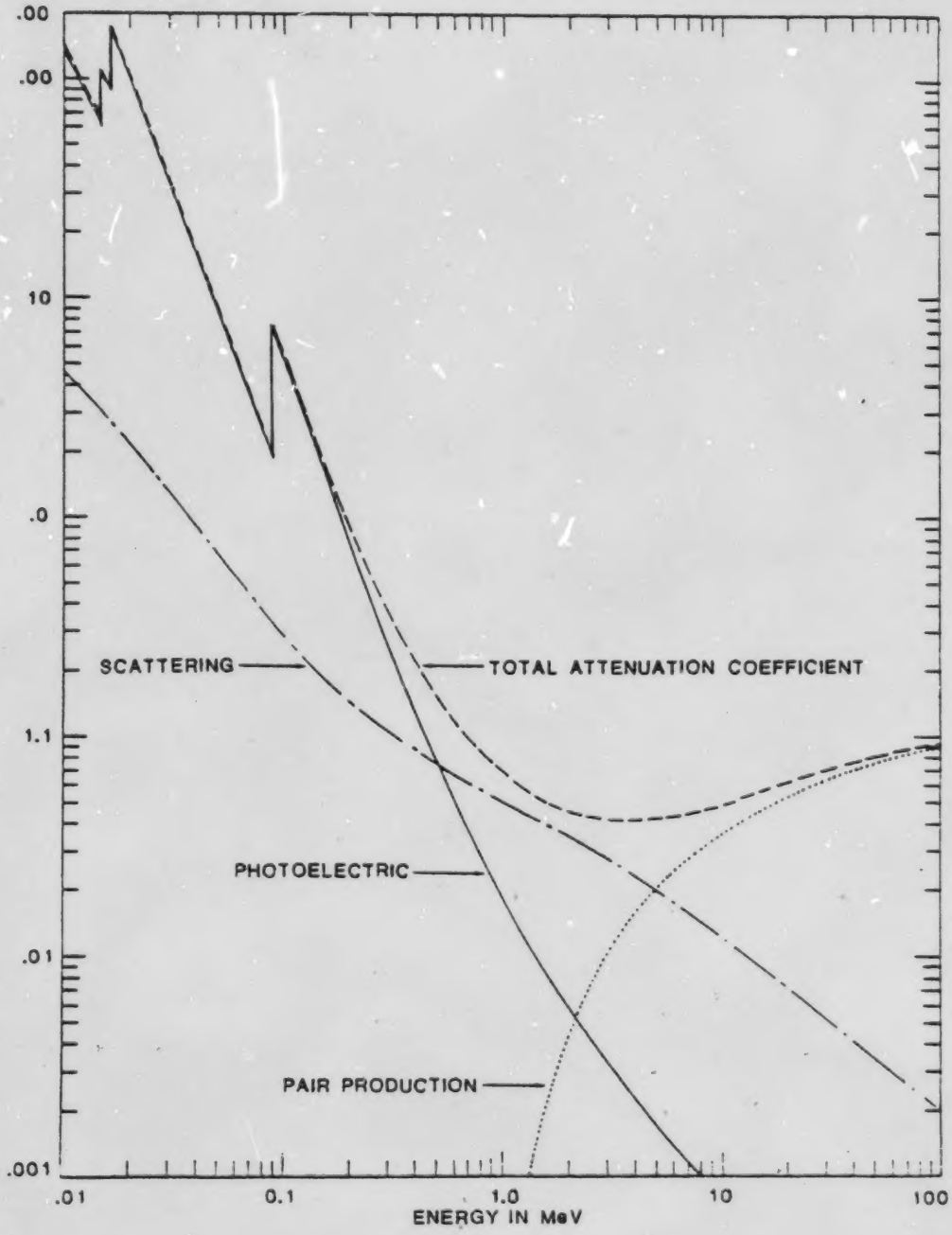


Figure 19-1-2. Mass attenuation coefficients for lead.

TABLE 19-1-1

MASS ATTENUATION COEFFICIENTS

Photon Energy	SiO_2	NaI	Air	Concrete	0.8N H_2SO_4	Bone	Muscle	Polystyrene	Lucite	Polyethylene	Bakelite	Pyrex Glass
keV												
10	19.0	139.	4.99	26.9	5.76	20.3	5.27	2.13	3.25	2.01	2.76	17.1
15	5.73	47.4	1.55	8.24	1.76	6.32	1.63	0.755	1.06	0.728	0.923	5.14
20	2.49	22.3	0.732	3.59	0.849	2.79	0.793	.424	0.551	.420	.492	2.25
30	0.859	7.45*	.349	1.19	.391	0.962	.373	.259	.298	.266	.277	0.786
40	.443	19.3	.248	0.605	.276	.512	.268	.217	.234	.228	.223	.431
50	.318	10.7	.208	.392	.231	.349	.227	.199	.208	.209	.200	.302
60	.252	6.62	.188	.293	.208	.274	.205	.188	.193	.198	.187	.242
80	.194	3.12	.167	.213	.185	.209	.183	.173	.176	.183	.171	.190
100	.169	1.72	.154	.179	.171	.180	.170	.163	.164	.172	.161	.166
150	.140	0.625	.136	.144	.150	.149	.149	.145	.146	.154	.143	.139
200	.126	.334	.123	.127	.137	.133	.134	.132	.133	.140	.130	.125
300	.108	.167	.107	.108	.118	.114	.118	.115	.115	.122	.113	.107
400	.0959	.117	.0954	.0963	.106	.102	.105	.103	.103	.109	.101	.0954
500	.0874	.0955	.0870	.0877	.0965	.0927	.0960	.0938	.0941	.0995	.0921	.0870
600	.0808	.0826	.0805	.0810	.0893	.0857	.0888	.0868	.0871	.0921	.0852	.0804
800	.0707	.0676	.0707	.0709	.0783	.0752	.0779	.0763	.0765	.0809	.0749	.0704
MeV												
1.0	.0636	.0586	.0636	.0637	.0704	.0676	.0700	.0685	.0687	.0727	.0673	.0633
1.5	.0518	.0469	.0518	.0519	.0573	.0550	.0570	.0558	.0559	.0592	.0548	.0516
2.0	.0447	.0413	.0445	.0448	.0492	.0473	.0489	.0478	.0480	.0507	.0470	.0444
3.0	.0363	.0366	.0358	.0365	.0396	.0383	.0393	.0383	.0385	.0405	.0377	.0361
4	.0317	.0351	.0308	.0319	.0340	.0331	.0337	.0327	.0329	.0345	.0322	.0314
5	.0287	.0346	.0275	.0290	.0303	.0297	.0300	.0290	.0292	.0305	.0286	.0282
6	.0266	.0347	.0252	.0270	.0277	.0274	.0274	.0263	.0266	.0277	.0260	.0263
8	.0241	.0353	.0223	.0245	.0243	.0244	.0240	.0228	.0232	.0239	.0227	.0237
10	.0226	.0368	.0204	.0231	.0222	.0226	.0219	.0206	.0211	.0215	.0206	.0222
15	.0209	.0402	.0181	.0215	.0194	.0204	.0192	.0176	.0182	.0182	.0178	.0204
20	.0203	.0433	.0170	.0210	.0182	.0194	.0179	.0162	.0168	.0166	.0164	.0198
30	.0202	.0484	.0162	.0210	.0172	.0189	.0168	.0149	.0157	.0151	.0153	.0195
40	.0204	.0520	.0161	.0213	.0169	.0189	.0165	.0144	.0153	.0145	.0148	.0198
50	.0208	.0548	.0161	.0218	.0168	.0190	.0166	.0142	.0151	.0142	.0147	.0201
60	.0212	.0571	.0162	.0222	.0169	.0193	.0165	.0142	.0151	.0141	.0147	.0204
80	.0218	.0605	.0165	.0229	.0171	.0197	.0167	.0142	.0152	.0141	.0148	.0210
100	.0224	.0629	.0168	.0235	.0174	.0201	.0170	.0144	.0154	.0142	.0150	.0215
150	.0234	.0670	.0174	.0247	.0180	.0210	.0175	.0147	.0159	.0145	.0154	.0225
200	.0241	.0695	.0179	.0254	.0184	.0215	.0179	.0150	.0162	.0147	.0157	.0232
300	.0250	.0724	.0185	.0264	.0190	.0223	.0185	.0155	.0167	.0152	.0162	.0240
400	.0256	.0741	.0189	.0269	.0194	.0228	.0189	.0158	.0171	.0155	.0166	.0245
500	.0260	.0752	.0192	.0273	.0197	.0231	.0192	.0160	.0173	.0157	.0168	.0249
600	.0262	.0760	.0194	.0276	.0199	.0233	.0194	.0162	.0175	.0159	.0170	.0252
800	.0266	.0771	.0197	.0281	.0202	.0237	.0197	.0165	.0178	.0161	.0173	.0256
GeV												
1	.0269	.0778	.0199	.0283	.0204	.0239	.0199	.0166	.0180	.0163	.0174	.0258
1.5	.0273	.0789	.0202	.0287	.0207	.0243	.0202	.0169	.0182	.0165	.0177	.0262
2	.0275	.0794	.0204	.0290	.0209	.0245	.0203	.0171	.0184	.0167	.0179	.0264
3	.0278	.0800	.0206	.0292	.0211	.0247	.0205	.0173	.0186	.0169	.0181	.0267
4	.0279	.0803	.0207	.0294	.0212	.0249	.0206	.0174	.0187	.0170	.0182	.0268
5	.0280	.0805	.0208	.0295	.0212	.0249	.0207	.0174	.0188	.0170	.0183	.0269
6	.0281	.0807	.0208	.0295	.0213	.0250	.0208	.0175	.0188	.0171	.0183	.0269
8	.0281	.0808	.0209	.0296	.0214	.0251	.0208	.0175	.0189	.0172	.0184	.0270
10	.0282	.0809	.0209	.0297	.0214	.0251	.0209	.0176	.0189	.0172	.0184	.0271
15	.0283	.0811	.0210	.0298	.0215	.0252	.0209	.0176	.0190	.0173	.0185	.0271
20	.0283	.0812	.0210	.0298	.0215	.0252	.0210	.0177	.0190	.0173	.0185	.0272
30	.0283	.0813	.0211	.0298	.0216	.0253	.0210	.0177	.0191	.0173	.0185	.0272
40	.0284	.0813	.0211	.0299	.0216	.0253	.0210	.0177	.0191	.0173	.0186	.0272
50	.0284	.0814	.0211	.0299	.0216	.0253	.0210	.0177	.0191	.0173	.0186	.0272
60	.0284	.0814	.0211	.0299	.0216	.0253	.0210	.0177	.0191	.0174	.0186	.0273
80	.0284	.0815	.0211	.0299	.0216	.0253	.0210	.0178	.0191	.0174	.0186	.0273
100	.0284	.0815	.0211	.0299	.0216	.0253	.0211	.0178	.0191	.0174	.0186	.0273

* K edge of Iodine--33.2keV 5.89. 30.9.

ENERGY OF RADIATION

The range of all types of radiation increases as energy increases. Alpha particles usually are emitted with an energy ranging from 4 to 10 MeV. The energy of beta particles ranges up to 3.15 MeV, although most have an energy of approximately 1 MeV. Gamma rays emitted by natural radioactive elements have energies from about 0.04 to 3.2 MeV. These radiation energies are reduced in intensity (attenuated) as the distance from their source increases by absorption of the particulate radiation by the air. Consequently, the energy of particulate radiation gradually is lost. The energy of electromagnetic radiation is merely reduced, since there is no particulate absorption involved in electromagnetic radiation.

INVERSE SQUARE LAW

Gamma-ray intensity obeys the inverse square law, so it varies inversely with the square of the distance from the source. This relationship allows intensity to be determined at varying distances, once the initial intensity at a specific distance is known. Equation 19-1-4 can be employed to determine intensity.

$$\frac{I_1}{I_2} = \frac{d_2^2}{d_1^2} \qquad \text{Eq. 19-1-4}$$

where

I_1 = the initial intensity at d_1.

d_1 = the initial distance.

I_2 = the desired intensity at d_2.

d_2 = the new distance.

It should be noted here that, because of absorption by the air of particulate radiation, the inverse square law describes particular radiation energy as a function of distance only when that radiation is occurring in the absence of air; i.e., in a vacuum.

EXAMPLE	
Given:	A gamma-ray point source with an exposure rate of 100 mR/hr at 2 meters.
Find:	The distance at which the exposure rate drops to 2.5 mR/hr.
Solution:	Use Equation 19-1-4:

$$\frac{I_1}{I_2} = \frac{d_2^2}{d_1^2}$$

$$\frac{100 \text{ mR/hr}}{2.5 \text{ mR/hr}} = \frac{d_2^2}{(2 \text{ m})^2}$$

Solve for d_2:

$$d_2^2 = \frac{4(100)}{2.5} = 160 \text{ m}^2$$

$$d^2 = 160 \text{ m}^2$$

$$d = 12.7 \text{ m.}$$

The exposure rate will drop to 2.5 mR/hr at 12.7 meters.

PHYSICAL PROPERTIES OF ABSORBING MEDIUM

The attenuation of radiation varies with the atomic number of the absorbing material. High atomic-number (high Z) materials, such as lead, attenuate radiation more effectively than materials with low atomic numbers. Generally, the denser the material, the better it will attenuate radiation.

HALF-VALUE LAYER

One measure of a material's ability to attenuate radiation is the thickness of its half-value layer (HVL). The HVL is the thickness of a material that will reduce the intensity of incident radiation to one-half of its original value (Figure 19-1-3). The intensity (I) of a beam of radiation passing through one half-value layer can be expressed as the following identity:

$$I = \tfrac{1}{2} I_0$$

where

I = intensity of radiation transmitted through a half-value layer of the absorber.

I_0 = initial intensity of the radiation.

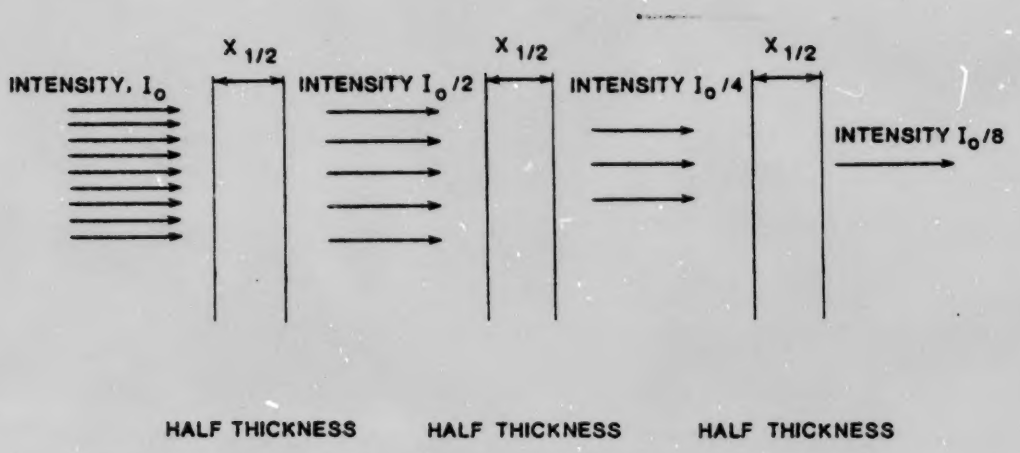


Figure 19-1-3. Half-value layer for attenuation of gamma rays.

HVL also can be defined in the following manner:

For 1 HVL, $I = \frac{1}{2} I_o$

use $I = I_o e^{-\mu x}$ and substitute

$$\frac{1}{2} I_o = I_o e^{-\mu x}$$

$$\frac{1}{2} = e^{-\mu x}$$

$$2 = e^{\mu x}$$

$$\ln 2 = \ln (e^{\mu x}) = \mu x$$

$$x = \frac{\ln 2}{\mu}$$

$$x = \frac{0.693}{\mu}$$

The half-value layer for a material can be calculated, therefore, by dividing the natural log of 2 (0.693) by the absorption coefficient. If the units of the absorption coefficient are cm^{-1}, the thickness is given in centimeters.

If thickness is given in units of g/cm^2, the mass attenuation coefficient is utilized; and the thickness is called the "density thickness" of the material.

The half-value layer concept is used frequently in the selection of attenuation materials that shield against specific electromagnetic radiations. Tables of HVL are available for various X-ray and gamma energies (Table 19-1-2). These tables frequently give values, not only for half-value layers, but also for tenth-value layers (TVL). A tenth-value layer is the thickness of absorber that will reduce the intensity of a beam of radiation to one-tenth of its original value. (See Figure 19-1-4.)

TABLE 19-1-2

HALF-VALUE AND TENTH-VALUE LAYERS

Approximate values obtained at high attenuation for the indicated peak voltage values under broad-beam conditions; with low attenuation these values will be significantly less.

Peak Voltage (kV)	Attenuation Material					
	Lead (mm)		Concrete (cm)		Iron (cm)	
	HVL	*TVL*	*HVL*	*TVL*	*HVL*	*TVL*
50	0.06	0.17	0.43	1.5		
70	0.17	0.52	0.84	2.8		
100	0.27	0.88	1.6	5.3		
125	0.28	0.93	2.0	6.6		
150	0.30	0.99	2.24	7.4		
200	0.52	1.7	2.5	8.4		
250	0.88	2.9	2.8	9.4		
300	1.47	4.8	3.1	10.4		
400	2.5	8.3	3.3	10.9		
500	3.6	11.9	3.6	11.7		
1,000	7.9	26	4.4	14.7		
2,000	12.5	42	6.4	21		
3,000	14.5	48.5	7.4	24.5		
4,000	16	53	8.8	29.2	2.7	9.1
6,000	16.9	56	10.4	34.5	3.0	9.9
8,000	16.9	56	11.4	37.8	3.1	10.3
10,000	16.6	55	11.9	39.6	3.2	10.5
Cesium-137	6.5	21.6	4.8	15.7	1.6	5.3
Cobalt-60	12	40	6.2	20.6	2.1	6.9
Radium	16.6	55	6.9	23.4	2.2	7.4

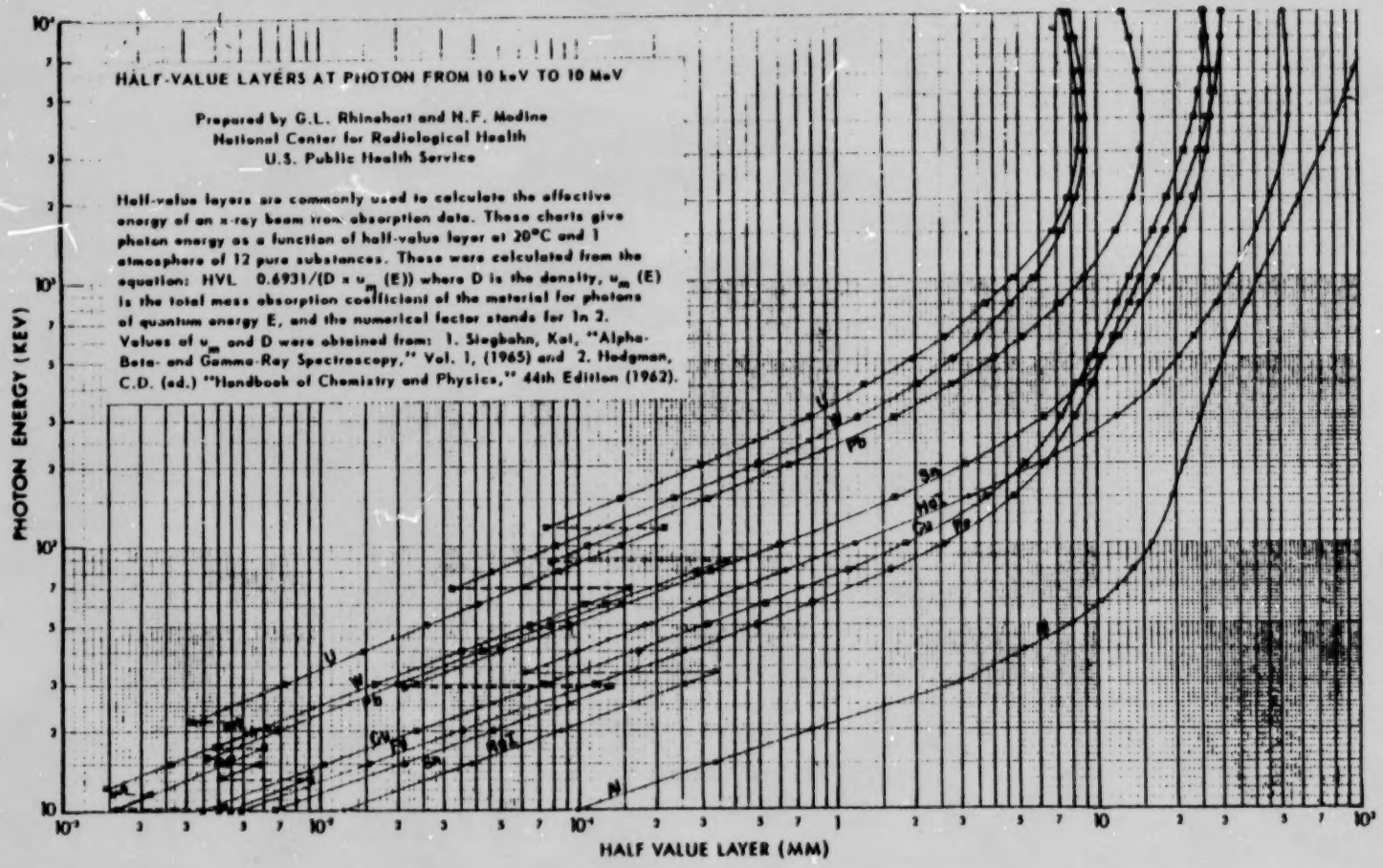

HALF-VALUE LAYERS AT PHOTON FROM 10 keV TO 10 MeV
Prepared by G.L. Rhinehart and N.F. Medina
National Center for Radiological Health
U.S. Public Health Service
Half-value layers are commonly used to calculate the effective energy of an x-ray beam from absorption data. These charts give photon energy as a function of half-value layer at 20°C and 1 atmosphere of 12 pure substances. These were calculated from the equation: HVL 0.6931/(D x μ_m (E)) where D is the density, μ_m (E) is the total mass absorption coefficient of the material for photons of quantum energy E, and the numerical factor stands for ln 2. Values of μ_m and D were obtained from: 1. Siegbahn, Kai, "Alpha-, Beta- and Gamma-Ray Spectroscopy," Vol. 1, (1965) and 2. Hodgman, C.D. (ed.) "Handbook of Chemistry and Physics," 44th Edition (1962).
PHOTON ENERGY (KEV)
HALF VALUE LAYER (MM)

Figure 19-1-4. Half-value layers as a function of photon energy for various absorbers.

In order to use the table, find the HVL of the radionuclide and of the absorber. Each additional HVL will reduce the exposure rate by one-half. Add HVLs until the exposure rate is reduced to the desired level.

EXAMPLE

Given: A ^{60}Co beam with an intensity of 96 mR/hr.

Find: The amount of lead shielding required to reduce the exposure rate to 2 mR/hr.

Solution: From Table 19-1-2 the HVL of ^{60}Co (lead) is 12 mm.

Set up a table:

Thickness (mm)	Fraction of Original Intensity (mR/hr)
0	96
12	48
24	24
36	12
48	6
60	3
72	1.5

Six HVL are required to reduce the exposure to a value less than 2 mR/hr.

After about 10 half-value layers, there are complications that alter the computations, but the method gives an approximation of the amount of shielding needed.

CROSS SECTION

One measure of the probability that a reaction will occur between incident radiation and a particle in the absorbing medium is the concept of cross section. This concept was discussed in Module 12-2, "Cross Sections and Reactions" (from Course 12, "Reactor Physics").

Cross section is expressed as the effective "area" that the nucleus presents for a reaction, and it is expressed in barns and is πr^2, where r is the radius of the particle. A barn is equal to 10^{-24} cm^2.

RADIATION INTERACTION AND DOSE EQUIVALENT

The concepts of specific ionization, linear energy transfer, quality factor, and dose equivalent have been discussed in previous modules of this series (specifically, Module 2-1 of Course 2), and they should be understood thoroughly.

Specific ionization is the number of ion pairs formed per unit distance traveled by the incident radiation. It decreases as the radiation energy increases and as particle mass decreases. Thus, low-energy beta particles have greater specific ionization than high-energy beta particles, and more massive alpha particles have greater specific ionization than beta particles. The greater a particle's specific ionization, the greater the danger it represents to biological tissue.

The linear energy transfer (LET) of incident radiation is the amount of energy deposited by the ionizing radiation per unit path-length in the absorbing medium. As specific

ionization increases, LET increases. It is expressed in units of keV per micron (10^{-6} m) and describes the energy loss per unit distance. Table 19-1-3 lists average LET values for different particles. The table indicates that the higher the rate of linear energy transfer of the radiation, the more damage it can do to an organism, due to deeper tissue penetration.

TABLE 19-1-3
AVERAGE LET VALUES

Particle	Mass (amu*)	Charge	Energy (kev)	Average LET (kev/micron)	Tissue penetration (microns)
Electron	0.00055	-1	1	12.3	0.01
			10	2.3	1
			100	0.42	180
			1000	0.25	5000
Proton	1	+1	100	90	3
			2000	16	90
			5000	8	350
			10000	4	1400
			200000	0.7	300000
Deuteron	2	+1	10000	6	700
			200000	1.0	190000
Alpha	4	+2	100	260	1
			5000	95	35
			200000	8	20000

Information pertaining to the specific ionization and LET of radiation can be used to formulate a relative measure of biological effectiveness. This relative measurement is called the "quality factor" (QF). Table 19-1-4 illustrates the relationship between QF and LET; Table 19-1-5 provides quality-factor values for various radiations.

TABLE 19-1-4

RELATIONSHIP BETWEEN QUALITY FACTOR AND LINEAR ENERGY TRANSFER

LET keV per micron in water	QF
3.5 or less	1
3.5 - 7.0	1-2
7.0 - 23	2-5
23 - 53	5-10
53 - 175	10-20

TABLE 19-1-5

QUALITY FACTOR VALUES FOR VARIOUS RADIATIONS

Radiation	QF
Gamma rays from radium in equilibrium with its decay products (filtered by 0.5-mm platinum)	1
X-rays	1
Beta rays and electrons of energy > 0.03 MeV	1
Beta rays and electrons of energy < 0.03 MeV	1.7
Thermal neutrons	3
Fast neutrons	10
Protons	10
Alpha rays	10
Heavy ions	20

The higher the LET, the higher the QF. QF is used to calculate dose equivalent (DE). DE is the unit of biological dose expressed in rems and is used for radiation safety purposes. The dose equivalent is determined by the following equation:

$$DE_{rems} = \text{dose (in rads)} \times QF \quad \text{Eq. 19-1-5}$$

The use of QF allowes DE to be determined for situations involving different types of radiation. DE for each radiation is calculated, and the totals are added.

EXAMPLE	
Given:	A person exposed for two hours to a fast neutron source with a dose rate of one mrad/hour, and exposed for 1/2 hour to a ^{60}Co source with an exposure rate of 4.8 mrad/hr.
Find:	Total dose equivalent.
Solution:	Obtain QF for sources from Table 19-1-5. QF for fast neutrons is 10. QF for ^{60}Co (a gamma-emitting source) is 1. Use Equation 19-1-5 for fast neutrons:
	DE_{rems} = dose x QF
	= 1 mrad/hr x 2 hr x 1o
	DE_{rems} = 20 mrem.
	For ^{60}Co:
	DE_{rems} = dose x QF
	= 4.8 mrad/hr x 1/2 hr x 1
	DE_{rems} = 2.4 mrem.
	Add DE for both sources:
	DE (fast neutrons) = 20 mrem
	DE (^{60}Co) = 2.4 mrem
	Total DE = 22.4 mrem

EXERCISES

1. The exposure rate from a ^{137}Cs gamma source is 500 mR/hr at a distance of six feet in air. At what distance does the exposure rate decrease to 10 mR/hr?
2. What thickness of lead will reduce a ^{60}Co beam to 1/5 of its initial intensity?
3. Using the HVL, find the thickness of concrete necessary to reduce an exposure rate of 1000 mR/hr to a rate of 5 mR/hr from a ^{137}Cs source.
4. A gamma-ray source has an exposure rate of 150 mR/hr at 1 meter in air. Find the distance at which the exposure rate will drop to 10 mR/hr.

MATERIALS

1 – Geiger-Mueller survey meter with side window GM tube. The GM tube should be used with the movable shield in place, covering the tube.
1 – Gamma source such as ^{60}Co with an activity of 5 mCi.
1 – Meter stick.
1 – Thin end window GM survey meter.
Two each of alpha, beta, and gamma check sources. Each type of source should have different energy.
Shielding material: lead sheets 1/16" thick, lucite, glass, aluminum, all 4" x 4".
1 – Scintillation dectector with scaler.
1 – Gamma-ray source such as ^{60}Co, with activity high enough to give several thousand counts-per-minute at a distance of about one foot.
15 – lead sheets, 3" x 3" square, approximately 1 mm thick.

Semilog graph paper for the Half-Value Layer experiment.

PROCEDURES

INVERSE SQUARE LAW

NOTE: Record data in Data Table 19-1-1.

1. Take a background measurement with no source present, and record the results.
2. Place the gamma source at the end of the meter stick.
3. Using the scale on the GM survey meter that gives about a 2/3 deflection of the meter, measure the exposure rate at 15 cm from the source. Record the measurement.
4. Repeat this procedure for source-to-detector distances of 30, 45, 60, 75, and 90 cm.
5. Make a graph of corrected count rate (measured count rate minus background) versus $1/d^2$, where d is the distance of the measurement.
6. Compare the theoretical exposure rates at these distances with the measured values, and explain any difference.

RADIATION ATTENUATION

NOTE: Record data in Data Table 19-1-2.

1. Take a background measurement with no source present.
2. Use the alpha standards, and measure activity with no shielding material. Record the results.

3. Measure the count rate from the alpha sources with each type of material in place, and record the results.
4. Repeat the measurements with the beta sources.
5. Repeat the measurements with the gamma sources and with the cap on the end of the tube.
6. Compare the degree of attenuation for each type of radiation and material. Does the density of the absorber affect attenuation? Compare the attenuation of the high-energy alpha with the low-energy alpha, of the high-energy beta with the low-energy beta, and of the high-energy gamma with the low-energy gamma.

HALF-VALUE LAYER

NOTE: Record data in Data Table 19-1-3.

1. Measure and record the background count rate.
2. Set up and maintain a constant source-to-detector distance.
3. Measure and record the count rate of the source with no lead shielding present.
4. Add one sheet of lead, measure and record the count rate.
5. Repeat the procedure until the count rate measured is less than one-half of the initial value (with no lead).
6. Make a plot on semilog graph paper of the corrected count rate versus the absorber thickness. Find the HVL. (Figure 19-1-4 shows the general shape of the curve.)
7. Determine the linear absorption coefficient, μ, and the mass attenuation coefficienct, μ_m. (The density

of lead is 11.35 g/cm^3.)

8. Compare these results with reference values, and discuss any differences.

DATA TABLES

DATA TABLE 19-1-1
INVERSE SQUARE LAW

Background _____ mRhr		
Distance (cm)	Count Rate (cpm)	Corrected Count Rate (cpm)
15		
30		
45		
60		
75		
90		

DATA TABLE 19-1-2
RADIATION ABSORPTION

Source _____	
Shield	Count Rate (cpm)
None	
Lucite	
Glass	
Aluminum	
Lead	

DATA TABLE 19-1-3
HVL MEASUREMENTS

Lead Absorber Thickness (mm)	Count Rate (cpm)	Corrected Count Rate (cpm)
0		

REFERENCE MATERIALS

Cember, H. Introduction to Health Physics. Elmsford, NY: Pergamon Press, 1969.

National Council on Radiation Protection and Measurements (NCRP). "Structural Shielding Design and Evaluation for Medical Use of X-rays and Gamma Rays of Energies up to 10 MeV," Report No. 49. Washington, DC: NCRP, 1976.

U.S. Department of Health, Education and Welfare. Radiological Health Handbook. Revised ed. Washington, DC: U.S. Government Printing Office, 1970.

1. List the three factors that influence radiation attenuation in matter.
   a. _____
   b. _____
   c. _____
2. Define the following terms:
   Linear absorption coefficient (μ).

   Mass attenuation coefficient (μ_m).

   Half-value layer (HVL).

   Tenth-value layer (TVL).

   Cross section.

   Barn.

Specific ionization.

Linear energy transfer (LET).

Quality factor (QF).

3. A gamma-ray source has an exposure rate of 500 mR/hr at 2 meters. Calculate the distance at which the exposure rate will drop to 300 mR/hr.

4. Calculate the amount of lead shielding required to reduce a ^{137}Cs beam with an intensity of 50 mR/hr to 2 mR/hr. Use Table 19-1-6.

TABLE 19-1-6

HALF-VALUE AND TENTH-VALUE LAYERS

Approximate values obtained at high attenuation for the indicated peak voltage values under broad-beam conditions; with low attenuation these values will be significantly less.

Peak Voltage (kV)	Attenuation Material					
	Lead (mm)		Concrete (cm)		Iron (cm)	
	HVL	TVL	HVL	TVL	HVL	TVL
50	0.06	0.17	0.43	1.5		
70	0.17	0.52	0.84	2.8		
100	0.27	0.88	1.6	5.3		
125	0.28	0.93	2.0	6.6		
150	0.30	0.99	2.24	7.4		
200	0.52	1.7	2.5	8.4		
250	0.88	2.9	2.8	9.4		
300	1.47	4.8	3.1	10.4		
400	2.5	8.3	3.3	10.9		
500	3.6	11.9	3.6	11.7		
1,000	7.9	26	4.4	14.7		
2,000	12.5	42	6.4	21		
3,000	14.5	48.5	7.4	24.5		
4,000	16	53	8.8	29.2	2.7	9.1
6,000	16.9	56	10.4	34.5	3.0	9.9
8,000	16.9	56	11.4	37.8	3.1	10.3
10,000	16.6	55	11.9	39.6	3.2	10.5
Cesium-137	6.5	21.6	4.8	15.7	1.6	5.3
Cobalt-60	12	40	6.2	20.6	2.1	6.9
Radium	16.6	55	6.9	23.4	2.2	7.4

5. Describe the relationship of specific ionization to mass, to energy, and to linear energy transfer.

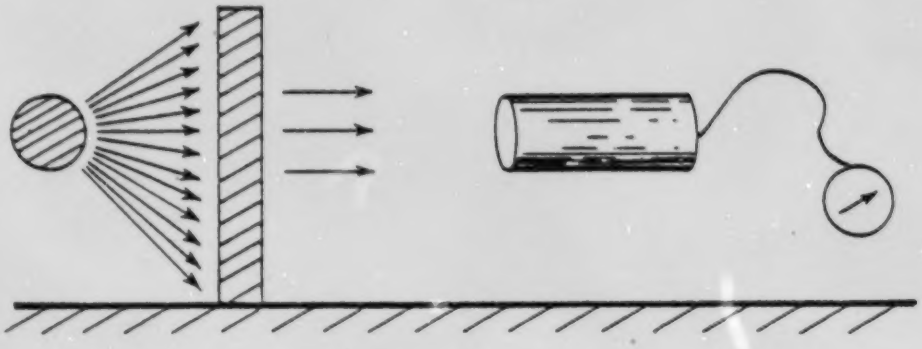
6. Describe the relationship of linear energy transfer to biological effectiveness and to quality factor.

7. The dose rate in a laboratory is 0.5 mrad/hr from thermal neutrons and 1.5 mrad/hr from gamma radiation. Based upon an eight-hour work day, how many days will pass before a person in the lab exceeds a maximum permissible dose equivalent of 1000 mrem?

NUCLEAR TECHNOLOGY

MODULE 19 - 2

BETA AND ALPHA PARTICLE ATTENUATION



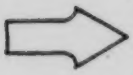
RADIATION SHIELDING

COURSE 19

ORD CENTER FOR OCCUPATIONAL RESEARCH AND DEVELOPMENT

COURSE MODULES

COURSE 19 RADIATION SHIELDING



MODULE 19-1 General Effect of Radiation Attenuation
MODULE 19-2 Beta and Alpha Particle Attenuation
MODULE 19-3 Neutron Attenuation
MODULE 19-4 Gamma Attenuation
MODULE 19-5 Background Radiation Reduction Techniques
MODULE 19-6 Backscattering and Wall Scattering of Radiation

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INTRODUCTION

BETA AND ALPHA PARTICLE ATTENUATION

One method of radiation protection uses shielding – a radiation-attenuating barrier placed between the source of radioactivity and the person or object to be protected. In theory, this barrier can be constructed of almost any material. Solids, such as aluminum and lead, often are used; liquids are used for some types of radiation; even air may be employed.

This module, which discusses radiation shielding and explores ways in which it absorbs alpha and beta radiation, demonstrates that the types and amounts of shielding needed depend upon the energy of the particle emitted and upon the atomic number of the absorbing material. The mechanisms by which energy is lost when a particle encounters a material are described.

MODULE PREREQUISITES

The student should have completed a course in basic algebra, and Course 1, "Radiation Physics," and Course 11, "Radiation Detection Systems."

OBJECTIVES

Upon completion of this module, the student should be able to:

1. Define:
   - Delta rays.
   - Specific ionization.
   - Bremsstrahlung.
   - Electron density (density thickness).
   - Inelastic collision.
   - Elastic collision.
   - Characteristic radiation.
2. Describe interactions of alpha particles with orbital electrons.
3. Describe interactions of beta particles with the nucleus and with orbital electrons.
4. Differentiate between alpha- and beta-particle absorption, from the standpoint of specific ionization and range.
5. Calculate degree of bremsstrahlung hazard from a beta source.
6. Determine dose rate for a given beta source.

ALPHA RADIATION

Alpha particles, which are relatively heavy, contain two protons and two neutrons per particle. Usually they are ejected as a result of disintegrations of radioactive nuclei of various heavy elements. The particles themselves are helium nuclei. Each has an atomic weight of 4.00277 AMU, and each carries two units of positive charge.

An alpha particle is emitted with an energy varying from 4 to 10 million electron volts (MeV) at a speed ranging from 1.39×10^9 to 2.19×10^9 cm/sec. Because of its size and relatively slow speed, the alpha particle travels only a few centimeters in air before being absorbed. Alpha radiation is the least penetrating of the three types (alpha, beta, gamma) and can be stopped by a sheet of paper.

Alpha radiation is an ionizing radiation that displaces electrons from atoms. When alpha particles interact with these atoms or molecules, ions are produced. About 35 electron volts are needed to produce an ion pair in air.

Alpha particles have a relatively large mass. Their paths through matter are, more or less, straight lines. When they interact with orbital electrons, most of their energy loss occurs as they impart kinetic energy to the electrons. The alpha particles themselves lose only a small amount of energy in each collision. Ionizing particles generated by the interaction of the alpha particle with matter are called "delta rays." These delta rays then will produce effects known as "secondary ionizing effects."

One measure of a particle's ability to interact with orbital electrons is its specific ionization (the number of ion pairs formed per unit length of path of a particle). The specific ionization of an alpha particle increases greatly at the end of its path (Figure 19-2-1).

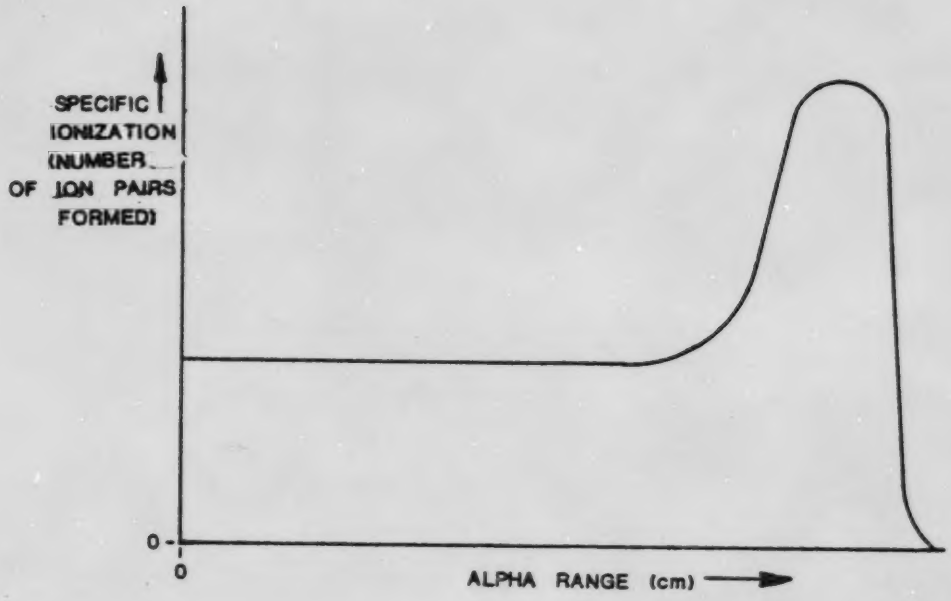


Figure 19-2-1. Specific ionization curve for alpha particles.

After many collisions with orbital electrons, the alpha particle loses a large amount of its kinetic energy. As its kinetic energy decreases, its velocity also decreases; therefore, it spends more time in the vicinity of a given atom. As this time increases, the probability of an interaction between the particle and an orbital electron becomes greater; and its specific ionization increases.

RANGE

A definite relationship exists between the energy of an alpha particle and its range (the distance from the emitting source to the point at which the last ion was produced).

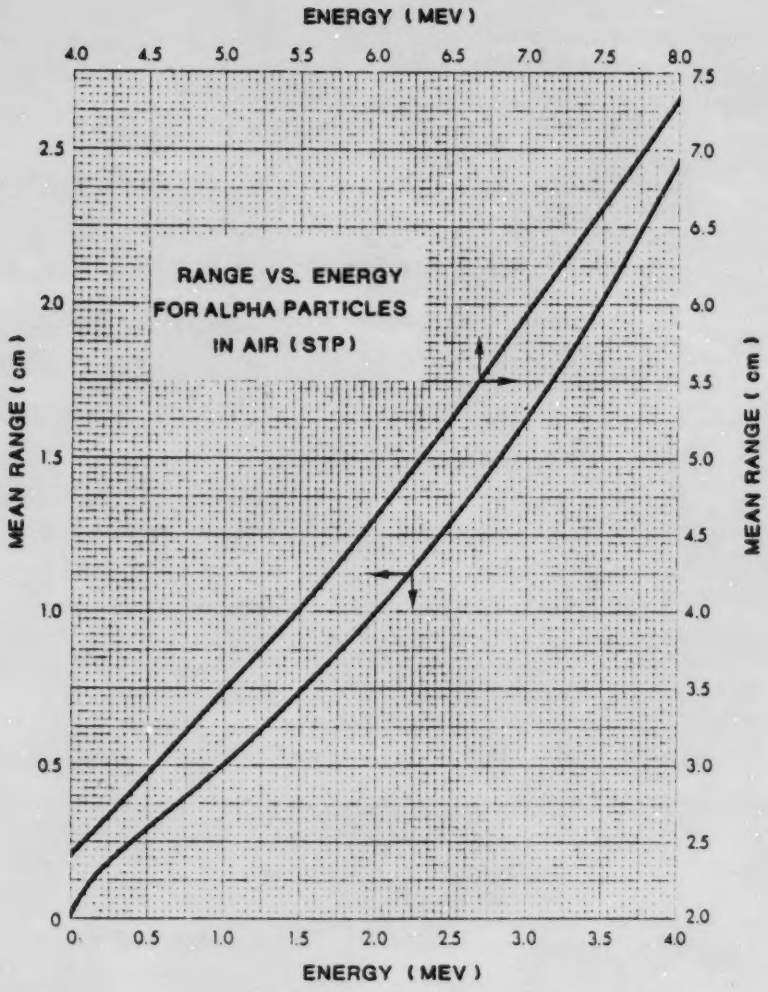


Figure 19-2-2. Range curve for alpha particle.

All alpha particles of the same energy have the same approximate range in matter. Figure 19-2-2 illustrates the relationship of range and energy of an alpha particle. On the figure, range of the alpha particle is defined as its "mean range"; that is, the range at which thickness of the absorber material (air, in this case) cuts the initial radiation

intensity to one-half its former strength.

The range of an alpha particle also is affected by the absorbing medium. The higher the atomic number of the absorbing medium, the shorter the particle's range. Because elements with higher atomic numbers have more electrons, an alpha particle traversing such an element has a greater chance of encountering electrons than it does in an element having a lower atomic number.

The range of an alpha particle in air can be approximated as follows:

$$\text{Range (R) in centimeters} = 0.56E(\text{MeV}) \quad \text{where } E < 4 \text{ MeV} \qquad \text{Equation 19-2-1 (Equation = Eq.)}$$

$$R\ (\text{cm}) = (1.24\ E[\text{MeV}]) - 2.62, \quad \text{where } 4 < E < 8 \text{ MeV} \qquad \text{Eq. 19-2-2}$$

where

R = range in cm of air at 1 atm and 15°C.

E = energy in MeV.

The range of an alpha particle in any other medium is given by Equation 19-2-3.

$$R_m\ (\text{mg/cm}^2) = 0.56\ A^{\frac{1}{3}}\ R \qquad \text{Eq. 19-2-3}$$

where

A = mass number of absorbing medium.

R = range of alpha in air (in cm).

R_m = range of alpha in medium (expressed in mg/cm^2).

EXAMPLE

Given: Energy of ^{238}U alpha is about 4.2 MeV.

Find: Thickness of aluminum necessary to stop alpha particles from ^{238}U.

Solution: $R_m\ (mg/cm^2) = 0.56\ A^{\frac{1}{3}}\ R$ (Eq. 19-2-3),

where

$$R\ (cm) = 1.24\ E(MeV) - 2.62$$
$$= (1.24)(4.2) - 2.62$$
$$= 2.59\ cm.$$
$$A = 27 \text{ for aluminum.}$$
$$R_m = (0.56)(27)^{\frac{1}{3}}(2.59)$$
$$R_m = 4.35\ mg/cm^2.$$

This value is expressed in cm as follows:

$$4.35 \times 10^{-3}\ g/cm^2 = (2.7\ g/cm^3)(t\ cm),$$

where

$2.7\ g/cm^3$ = density of aluminum.

Thickness t:

$$t = \frac{4.35 \times 10^{-3}\ \frac{g}{cm^2}}{2.7\ g/cm^3}$$

$$t = 1.61 \times 10^{-3}\ cm = 0.00161\ cm.$$

SHIELDING CONSIDERATIONS

Because of the short range of an alpha particle and its quick absorption in the outermost layers of human tissue, its external radiation hazard is not very great. Adequate shielding can be obtained from glass or metal walls of a container, or from air as an absorbing material.

BETA RADIATION

Beta particles originate in the nucleus of an atom. Once outside the nucleus they become very high-speed electrons. Each has a range of several feet in air and can be stopped by a thin sheet of aluminum or several sheets of paper. A beta particle interacts with nuclei and with orbital electrons. When a beta particle interacts with a nucleus or orbital electron, it is deflected from its path.

There are two kinds of nuclear interactions – elastic and inelastic collisions. In an elastic collision, the kinetic energy of the whole system (beta particle and atom) remains unchanged. The beta-particle energy before the collision equals the beta-particle energy after the collision. This effect, called "Rutherford Scattering," is illustrated in Figure 19-2-3.

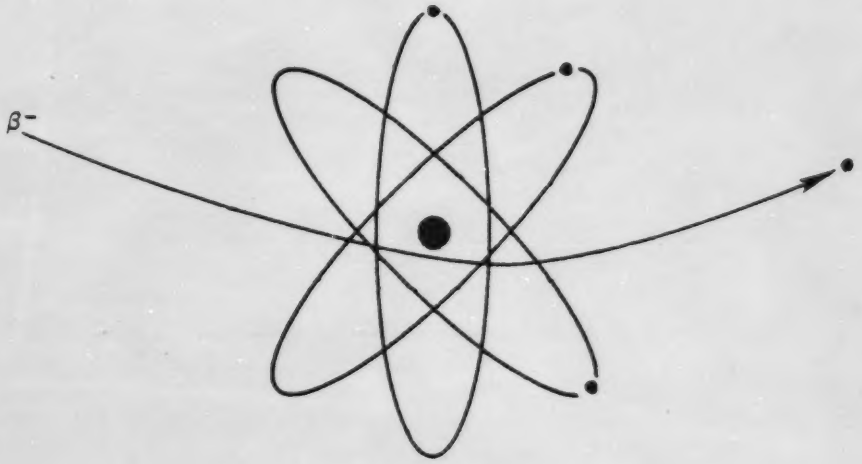


Figure 19-2-3. Rutherford scattering.

In an inelastic collision, the kinetic energy of the system decreases. The beta particle is slowed down by the nucleus. As it slows, it emits electromagnetic X-ray radiation. The decrease in kinetic energy of the system is equal to the energy of the X-ray. This loss of energy, resulting in X-ray emission, is called "bremsstrahlung" (Figure 19-2-4). (In German, this term means "braking radiation.")

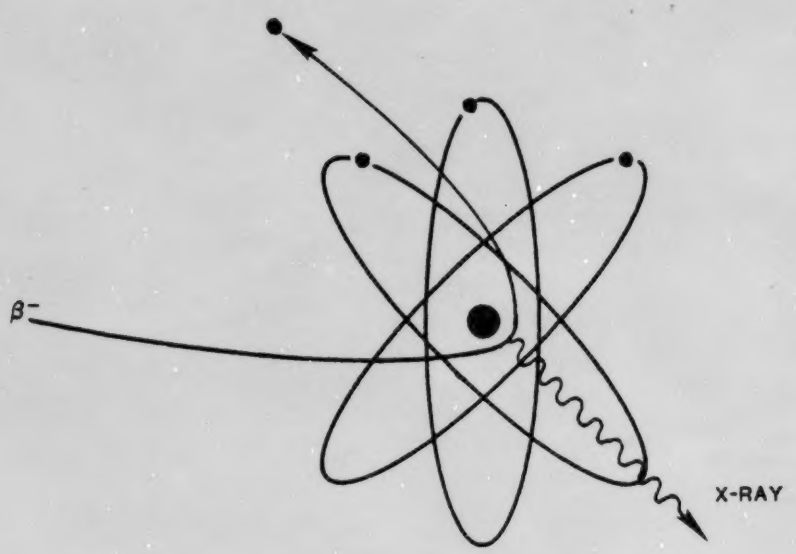


Figure 19-2-4. Bremsstrahlung.

Most beta-particle interactions are with orbital electrons. Because the beta particle has a negative charge, it repulses the electron, causing it to dissociate from the atom (Figure 19-2-5). The final energy of a beta particle equals its initial energy less the binding energy

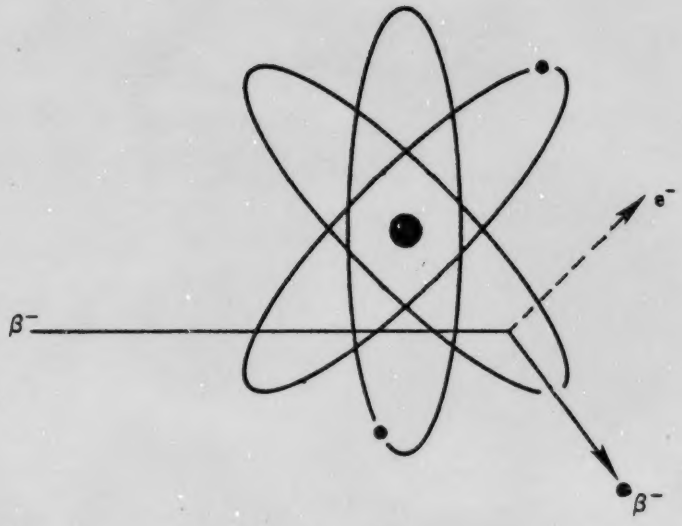


Figure 19-2-5. Ionization.

and kinetic energy imparted to each electron it encounters, as described by Equation 19-2-4.

$$E_f = E_i - (\phi + \tfrac{1}{2} mv^2) \qquad \text{Eq. 19-2-4}$$

where

ϕ = electron binding energy in MeV.
$\frac{1}{2} mv^2$ = electron kinetic energy in MeV.
E_f = final energy.
E_i = initial energy.

If the electron comes from a K, L, or M orbital band, an X-ray is emitted as the atom returns to its normal energy level. This X-radiation is called "characteristic radiation." The electrons (delta rays) themselves cause secondary radiation.

X-radiation is very penetrating. The degree of bremsstrahlung hazard from a given source can be estimated by the following equation:

$$f = 3.5 \times 10^{-4} ZE \qquad \text{Eq. 19-2-5}$$

where

f = fraction of incident beta-particle energy converted to X-ray.
Z = atomic number of absorbing medium.
E = maximum energy of beta particle in MeV.

SPECIFIC IONIZATION

Because of the difference in particle mass and velocity, specific ionization for a beta particle is much less than for an alpha particle. Also, because specific ionization is inversely related to particle velocity, and because beta particles travel close to the speed of light, the specific ionization of a beta particle is low.

Figure 19-2-6 (specific ionization versus beta-particle energy) demonstrates that, as energy increases, ionization decreases. Specific ionization reaches a minimum near 1 MeV. At energies above this value, specific ionization

increases slightly because of relativistic considerations.

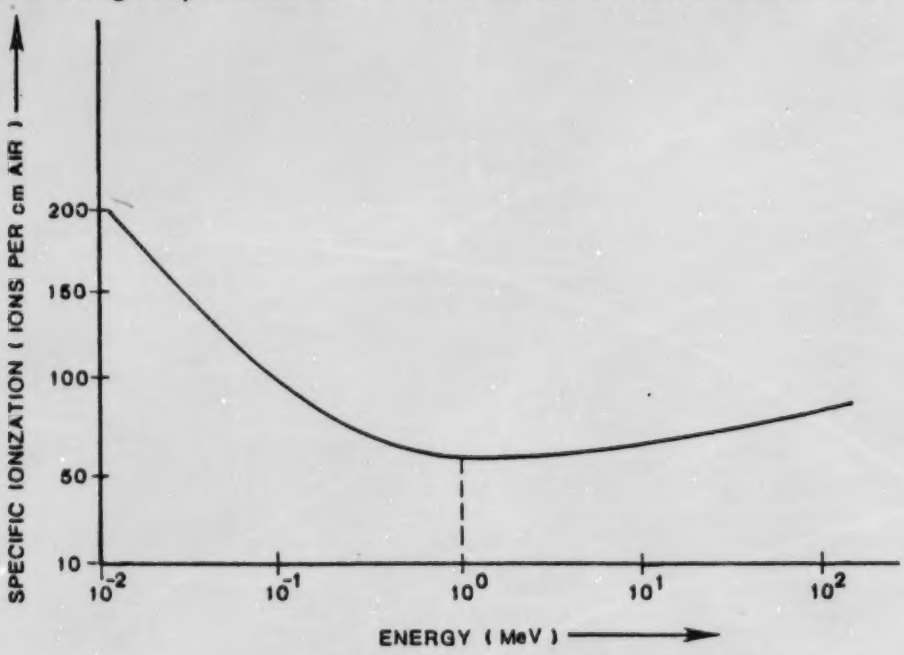


Figure 19-2-6. Specific ionization curve for beta particles.

RANGE

Figure 19-2-7 gives the absorption curve for beta particles. When absorber thickness is increased, intensity (of a beam of beta particles) decreases rapidly until it reaches a minimum value. This point of maximum value on the absorption curve is the "range" of the beta particle in the absorbing medium.

Much of the absorption curve is a straight line, which indicates an approximate exponential relationship. It is not a true exponential relationship, however, because the beta particle has a finite range.

The beta absorption curve does not go to zero intensity; the residual intensity is due to bremsstrahlung.

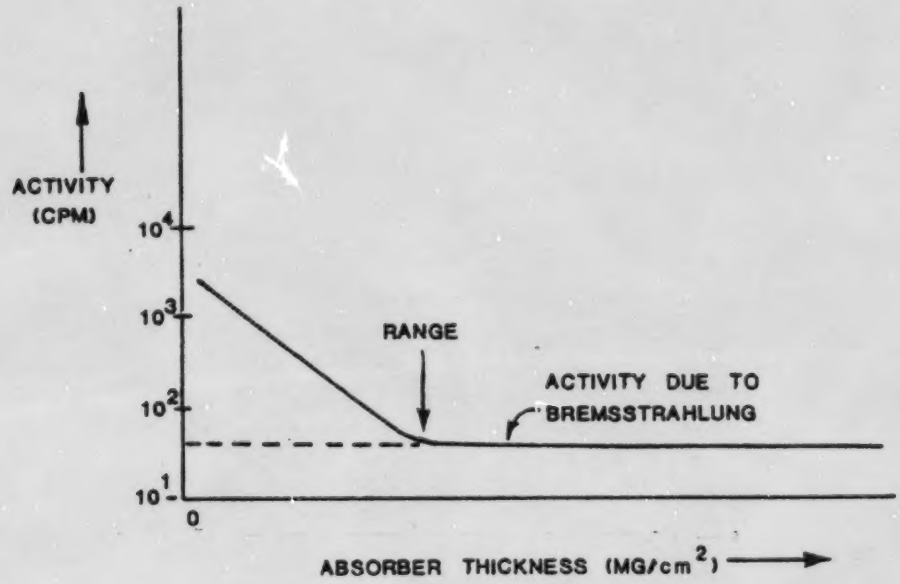


Figure 19-2-7. Absorption curve for beta particles.

In Figure 19-2-8, the relationship between beta energy and range is given for four different absorbing mediums. The graph indicates two things:

- In a given medium, the beta range increases as beta-particle energy increases.
- For a given energy, the range is inversely proportional to absorber density.

Beta-particle absorption is primarily a function of the electron density (also called "density thickness") of the absorbing medium. The electron density is approximately proportional to the density of the material. Equation 19-2-6 defines the density thickness of an absorbing medium.

$$t_d\ (g/cm^2) = \rho\ (g/cm^3) \cdot t_l\ (cm) \qquad \text{Eq. 19-2-6}$$

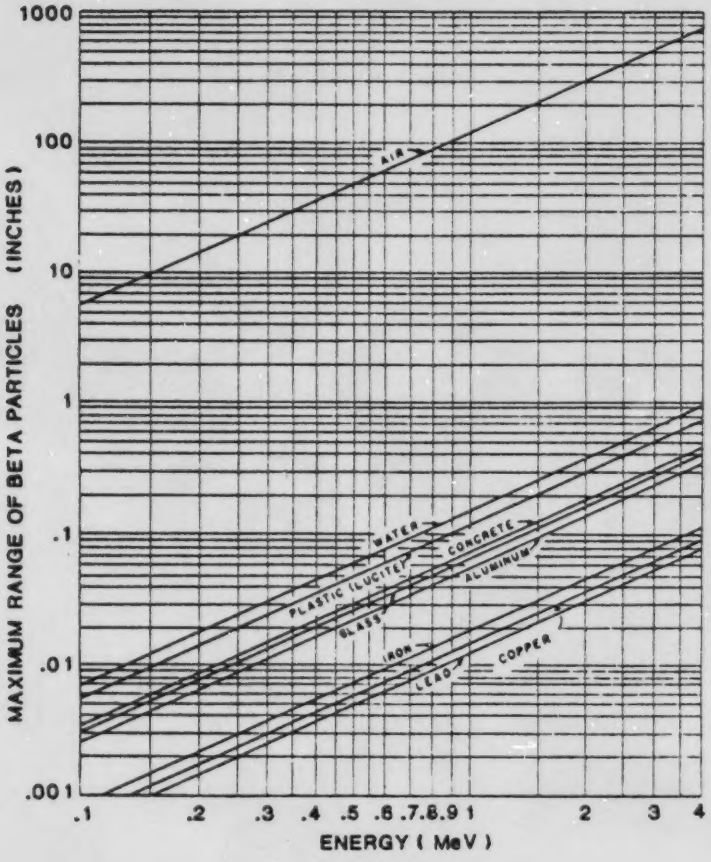


Figure 19-2-8. Range curve for beta particles of different energies.

where

ρ = absorber density in g/cm^3.

t_l = thickness of the absorber in cm.

t_d = density thickness in g/cm^2.

This equation allows the absorber to be specified in units of g/cm^2.

When density thickness is known, a range-energy curve can be constructed for beta particles (Figure 19-2-9). This curve can be used to identify unknown beta emitters. The range of a beta particle is measured and its corresponding energy calculated. These two findings then are matched to a table that compares range and energy of beta particles with corresponding beta emitters.

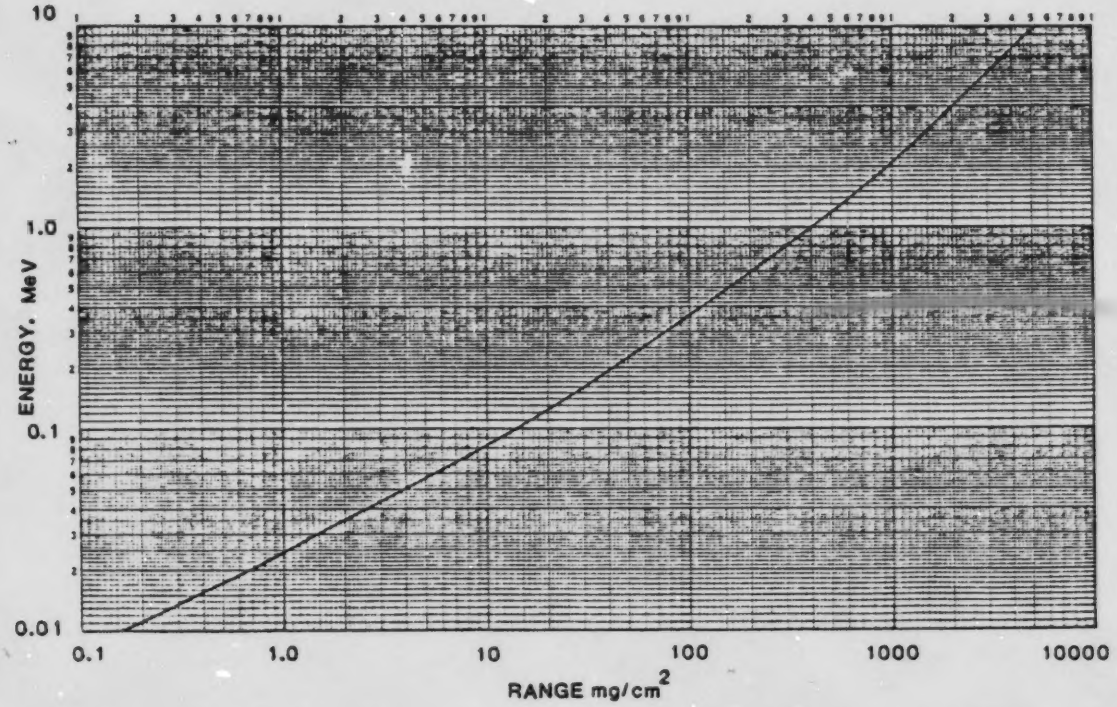


Figure 19-2-9. Range energy curve for beta particles.

SHIELDING FOR BETA RADIATION

Although the beta-particle range is greater than the range of alpha particles, air still can be used as an absorbing medium. The glass or metal walls of a beta container usually can absorb all beta particles.

X-rays that result from bremsstrahlung do create additional shielding problems. The production of bremsstrahlung is a function of the atomic number of the absorbing medium (the lower the atomic number, the fewer X-rays created); therefore, a material with a low atomic number (low Z) often is used as shielding. The best beta shield has an inner wall of low-Z material (such as plastic) to minimize bremsstrahlung production and has a high-Z outer wall (such as lead) to absorb any X-rays created.

DETERMINING EXTERNAL DOSE

The dose from a beta emitter is given by Equation 19-2-7.

$$D = E \times A \times k \times G \times P \times Q \times T \times f \qquad \text{Eq. 19-2-7}$$

where

D = dose rate in rems per second.
E = average beta energy in MeV = $\frac{1}{3}E_{max}$.
A = source activity in curies.
k = conversion factors - 1.6×10^{-8} g-rads/MeV.
G = geometry factor; for a point source.

$G = \frac{A}{4\pi d^2}$ where d is the distance from the source in cm, and A is the area of the detector window in cm^2.

P = fraction of energy absorbed in a depth of 1 g/cm^2 = 1/Range.

Q = quality factor of the radiation.

T = time of exposure in seconds.

f = fraction of beta particles emitted per disintegration.

This equation does not take into consideration attenuation in air, scatter, or self-absorption.

EXAMPLE

Given: 100 mCi ^{90}Sr - ^{90}Y point source.

Find: Dose rate at a point 3" from a source.

Solution: In equilibrium, the activity of ^{90}Sr = the activity of ^{90}Y, and the total dose rate is the sum of the individual dose rates. From Equation 19-2-7 –

$$D = E \times A \times k \times G \times P \times Q \times T \times f.$$

For ^{90}Sr, the values for Equation 19-2-7 are these:

$$E = \frac{1}{3}\left(0.546 \frac{MeV}{dis}\right) = \frac{0.182\ MeV}{dis}$$

$$A = 0.1\ Ci.$$

$$k = \frac{1.6 \times 10^{-8}\ g\ rads}{MeV}$$

$$G = \frac{1}{4\pi\ (3\ in)^2} = \frac{1}{4\pi(7.62\ cm)^2}$$

$$P = \frac{1}{\text{Range}} = \frac{1}{0.175\ g/cm^2} = 5.71\ \frac{cm^2}{g}$$

$$Q = 1\ \frac{rem}{rad}$$

$$T = \left(\frac{3.7 \times 10^{10}\ dis}{sec\ Ci}\right)$$

$$f = 1$$

$$D_{Sr} = \left(\frac{0.182\ MeV}{dis}\right)(0.1\ Ci)\left(\frac{1.6 \times 10^{-8}\ g\ rads}{MeV}\right)$$

$$\left(\frac{1}{4\pi\ (7.62\ cm)^2}\right)\left(5.71\ \frac{cm^2}{g}\right)\left(1\ \frac{rem}{rad}\right)$$

$$\left(\frac{3.7 \times 10^{10}\ dis/sec}{Ci}\right)(1)$$

$$= 8.43 \times 10^{-2}\ rem/sec$$

The dose rate for $^{90}Sr = 3.03 \times 10^2$ rem/hr. If the same equation is used for ^{90}Y, the E and P values are calculated thusly:

$$E = \frac{1}{3}\left(\frac{2.27\ MeV}{dis}\right) = \left(\frac{0.756\ MeV}{dis}\right)$$

$$P = \frac{1}{1.1}\ g/cm^2 = \frac{0.909\ cm^2}{g}$$

Substitute these values into the equation:

$$\left(\frac{0.756\ MeV}{dis}\right)(0.1\ Ci)\left(\frac{1.6 \times 10^{-8}\ g\ rads}{MeV}\right)$$

$$\left(\frac{1}{4\pi\ (7.62\ cm)^2}\right)\left(\frac{0.909\ cm^2}{g}\right)\left(1\ \frac{rem}{rad}\right)$$

$$\left(\frac{3.7 \times 10^{10}\ dis}{sec\ Ci}\right)(1) = 5.58 \times 10^{-2}\ rem/sec.$$

Dose rate for $^{90}Y = 2.01 \times 10^2$ rem/hr.

Total dose rate = $D_{^{90}S} + D_{^{90}Y}$

$= 3.03 \times 10^2$ rem/hr +

2.01×10^2 rem/hr

$= 5.04 \times 10^2$ rem/hr.

EXERCISES

1. Calculate the range of a 5-MeV alpha particle in aluminum.
2. Calculate the percent of energy dissipated as bremsstrahlung by 1-MeV beta particles as they pass through aluminum.
3. Calculate the dose rate at a point 12 inches from a 50-mCi ^{32}P point source. (Ignore air attenuation, scatter, and self-absorption.)

MATERIALS

Laboratory scaler with end window GM detector in sample holder.
Scintillation detector.
^{32}P source.
Sheets of lead – 1/16-inch thick; 2 1/2-inches diameter.
Sheets of plastic – 1/8-inch thick; 2 1/2-inches diameter.

PROCEDURES

BREMSSTRAHLUNG PRODUCTION

NOTE: Record data in Data Table 19-2-1.

1. Place ^{32}P source in GM detector sample holder in such a manner that a count rate of several thousand per minute is obtained. (See Figure 19-2-10 for illustrated lab setup.)

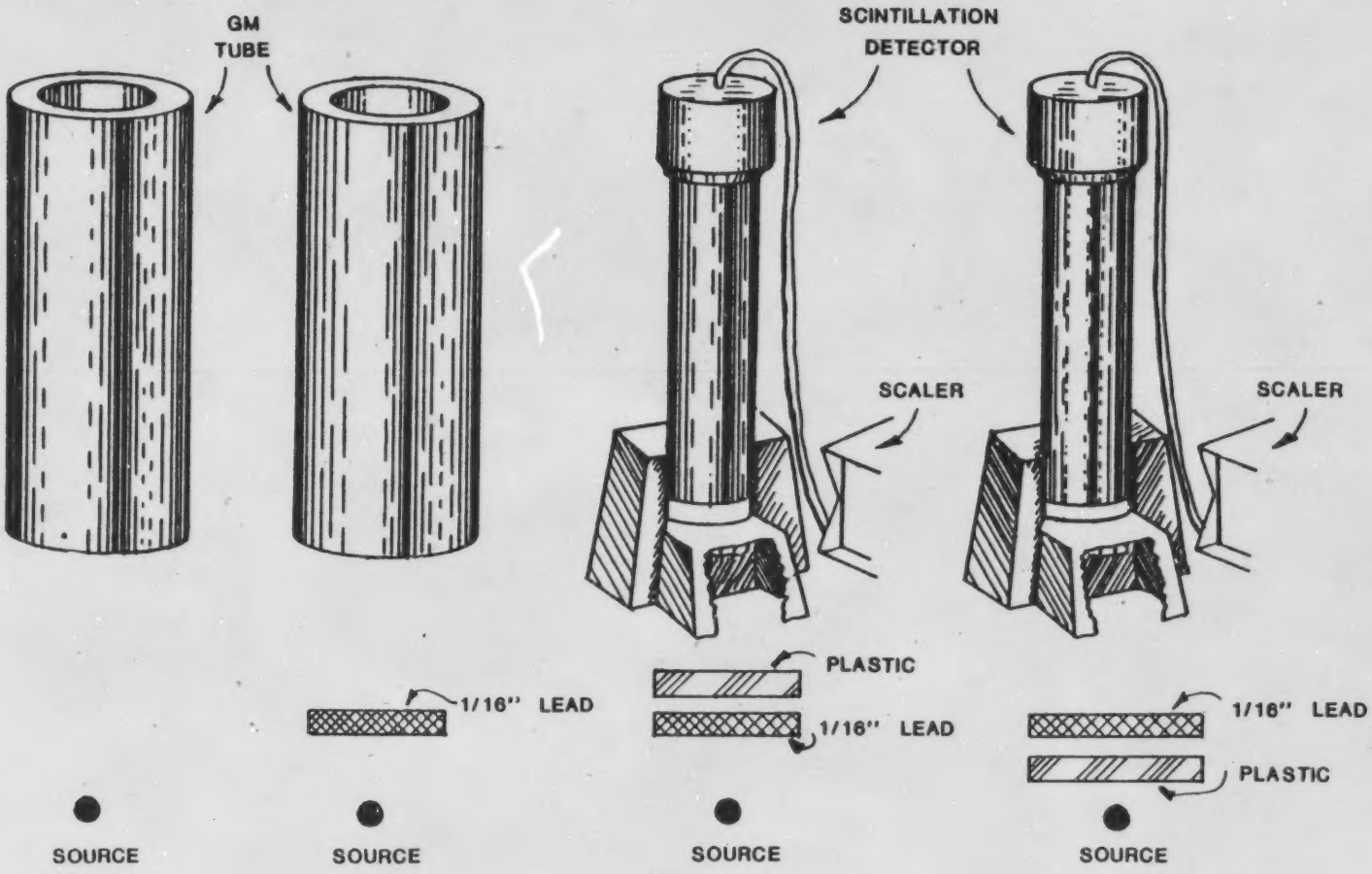


Figure 19-2-10. Illustrated lab setup for bremsstrahlung production.

2. Cover ^{32}P source with 1/16-inch lead; observe decrease in count rate (even when GM tube is brought close to source).
3. Use gamma-sensitive scintillator; observe count rate with lead in place. (The increase in count rate is due to bremsstrahlung.)
4. With lead in place between source and scintillation detector, measure activity; then place 1/8-inch sheet of plastic between source and lead; record activity. Next, place plastic between lead and detector; record activity. Explain difference in count rates.

Exp. No.	Count Rate, No Shield	Count Rate, 1/16" Lead	Gamma-Sensitive 1/16" Lead	Increase Count Rate	Source Lead Count	Source Plastic Lead Count	Source Lead Plastic Count
1							
2							
3							
4							
5							
Total =							
Average Count Rate = Total 5/=							

DATA TABLE 19-2-1

Cember, H. Introduction to Health Physics. Oxford, England: Pergamon Press, 1969.

Chase, G.D. and Rabinowitz, J.L. Principles of Radioisotope Methodology. 3rd ed. Minneapolis: Burgess Publishing Co., 1967.

Friedlander, G. and Kennedy, J. Nuclear and Radiochemistry. New York: John Wiley and Sons, 1949.

U.S. Department of Health, Education and Welfare. Radiological Health Handbook. Washington, DC: U.S. Government Printing Office, 1970.

1. Define the following terms:

   Delta ray.

   Specific ionization.

   Bremsstrahlung.

   Density thickness.

   Inelastic collision.

   Elastic collision.

   Characteristic radiation.

2. Describe the different types of interactions between alpha particles and orbital electrons.

3. Describe the different types of interactions between beta particles and (1) nuclei and (2) orbital electrons.

4. Explain differences between alpha- and beta-particle absorption from the standpoint of specific ionization and range.

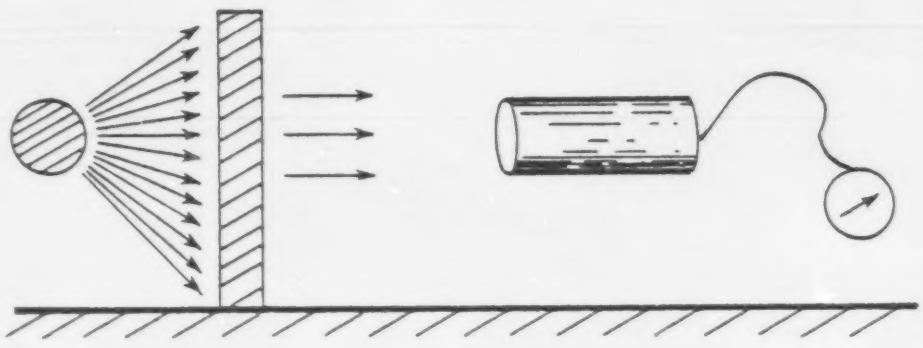
5. Explain how the degree of bremsstrahlung hazard from a beta source depends upon the atomic number of an absorber.

6. Calculate dose rate 4" from a 500-mCi ^{35}S point source. (Ignore air attenuation, scatter, and self-absorption.)

NUCLEAR TECHNOLOGY

MODULE 19 - 3

NEUTRON ATTENUATION



RADIATION SHIELDING

COURSE 19

CENTER FOR OCCUPATIONAL RESEARCH AND DEVELOPMENT

COURSE MODULES

COURSE 19 RADIATION SHIELDING

MODULE 19-1 General Effect of Radiation Attenuation
MODULE 19-2 Beta and Alpha Particle Attenuation
MODULE 19-3 Neutron Attenuation
MODULE 19-4 Gamma Attenuation
MODULE 19-5 Background Radiation Reduction Techniques
MODULE 19-6 Backscattering and Wall Scattering of Radiation

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INTRODUCTION

Neutrons are massive, uncharged particles. Because of their lack of charge, they interact with the nucleus of atoms rather than with orbital electrons. These interactions result in the induction of radioactivity.

This module discusses neutron sources, the interaction of neutrons with nuclei, and the attenuation of neutrons in matter. Calculations of neutron flux and attenuation, shielding requirements, and external neutron dose are included.

MODULE PREREQUISITES

The student should have completed Course 1, "Radiation Physics"; Course 2, "Radiation Protection I"; Course 17, "Radiation Protection II"; and one year of high school algebra.

OBJECTIVES

Upon completion of this module, the student should be able to:

1. List three radionuclide neutron sources, and describe their characteristics.
2. List and describe energy classifications of neutrons.
3. Describe neutron interaction with the nucleus.

4. Describe the relationship of absorption cross section and neutron energy.
5. Given the cross section for a specific neutron energy in the 1/V region, calculate the cross section for another energy within this range.
6. Given all necessary variables, calculate neutron flux, neutron attenuation, shielding requirements, and external neutron dose.

NEUTRON SOURCES

Neutron sources can be created by spontaneous fission and by nuclear bombardment. Sources created by nuclear bombardment can be divided into two broad categories:

- Radionuclide sources.
- Machine sources.

RADIONUCLIDE SOURCES

The most common neutron source is the alpha-neutron source, created by the bombarding of elements of low atomic number, such as beryllium with alpha particles. Alpha particle energy must be greater than 3.7 MeV. Equation 19-3-1 illustrates a typical reaction.

$$^{9}_{4}Be + ^{4}_{2}He \longrightarrow (^{13}_{6}C) \longrightarrow ^{12}_{6}C + ^{1}_{0}n + 5.65\ MeV$$ Equation 19-3-1

(Equation = Eq.)

Some common alpha sources are polonium-210, radium-226, plutonium-239, and americium-241. Table 19-3-1 lists some typical alpha-neutron sources.

TABLE 19-3-1

CHARACTERISTICS OF SOME IMPORTANT (α,n) SOURCES

Sources	Half-life	Maximum Neutron Energy	Average Neutron Energy	Yield	Remarks
		Mev	Mev	n/secX10^{-6} curie	
Po^{210}-Li	138.40d	1.52	0.48	.05	
Po^{210}-Be	138.40d	10.87	4.2	2.5	
*RaDEF-Be	19.4y	10.87	4.5	2.5	Po-Be with a long half-life.
*Ra-Be	1622y	13.08	* 5.9	15	
Em^{222}-Be	3.825d	13.08		15	
Pu^{239}-Be	24,400y	10.74	4.5	0.061 (per g)	
Ac^{227}-Be	21.8y	12.79	4.6		Made by irradiating radium in reactor.
Po^{236}-Be	2.93y	10.71			
*RaBeF	1622y	13.08		2.53	Proposed std source.
Po^{210}-B	138.40d	B^{10} 6.29, B^{11} 4.48.		0.6	Relatively monoenergetic.
*Ra-B	1622y	B^{10} 8.58, B^{11} 7.25.			
Po^{210}-F	138.40d	2.8	1.4	0.2	
Po^{210}-Na	138.40d	4.45		0.04	
Am^{241}-Be	462y				Suggested for stoichiometric std source.
Cm^{242}-Be	162.5d				
Mock fission[b]	138.40d	10.87	1.6	0.4	

*High rate of gamma emission occurs when radium sources are used.

Neutron emission also can be created by surrounding a gamma emitter with either deuterium or beryllium. The resultant reaction is described by Equation 19-3-2.

$$Be_4^9 + \gamma \longrightarrow Be_4^8 + n_0^1 - 1.666 \text{ MeV} \qquad \text{Eq. 19-3-2}$$

Deuterium and beryllium are used since their binding energies are low (2.23 MeV for deuterium, and 1.67 MeV for beryllium), and since the highest gamma-ray energy emitted by a suitable nuclide is 4 MeV. The neutrons emitted by these sources have energies less than 1 MeV (Table 19-3-2).

TABLE 19-3-2

CHARACTERISTICS OF SOME IMPORTANT (γ,n) SOURCES

Sources	Half-life	E_γ	E_n
		Mev	Mev
Na^{24}+Be	14.8h	2.76	0.83
Na^{24}+D_2O	14.8h	2.76	0.22
Ga^{72}+Be	14.1h	1.87, 2.21, 2.51	(0.78)
Y^{88}+Be	87d	1.9, 2.8	0.158±0.005
In^{116}+Be	54m	1.8, 2.1	0.30
Sb^{124}+Be	60d	1.7	0.024±0.003
La^{140}+Be	40d	2.50	0.62
RdTh+D_2O	1.90y	2.62 (ThC")	0.197±0.010
MsTh+Be	6.7y	1.80, 2.62	0.827±0.300
MsTH+D_2O	6.7y	2.62 (ThC")	0.197±0.010
Ra+Be	1622y	1.69, 1.75, 1.82, 2.09, 2.20, 2.42.	0.7 max

Spontaneous fission (SF) sources are heavy elements, artificially produced, that undergo fission as a mode of radioactive decay. The fission is accompanied by emission of two or three neutrons and several gamma rays. SF sources are radioactive in the common sense and also undergo alpha decay. All these sources have relatively short half-lives and only one, californium-252, has a half-life long enough to make it useful. SF sources are usually expensive and

difficult to obtain. Table 19-3-3 lists the characteristics of some SF neutron sources. Notice that sources with a low number of alphas per fission are "purer" and, hence, more useful neutron sources.

TABLE 19-3-3

CHARACTERISTICS OF SOME SPONTANEOUS FISSION NEUTRON SOURCES

Nuclide	Half-life (SF)	Half-life (α decay)	Alphas per fission*	Neutrons per fission	Neutrons per g sec
U^{232}	8×10^{13}y	74y	1.1×10^{12} 6.5×10^{12}, after aging with 1.9 yr half-life		
Pu^{236}	3.5×10^{9}y	2.7y	1.3×10^{3}	1.9	3.1×10^{4}
U^{238}	8.3×10^{15}y	4.51×10^{9}y	1.8×10^{6}		
Pu^{238}	4.9×10^{10}y	89.6y	5.5×10^{8}	2.0	2.3×10^{3}
Pu^{240}	1.3×10^{11}y	6600y	1.9×10^{7}	2.1	7.0×10^{2}
Pu^{242}	7.2×10^{10}y	3.8×10^{5}y	1.9×10^{5}	2.3	
Cm^{242}	7.2×10^{6}y	162.5d	1.6×10^{7}	2.3	1.8×10^{3}
Cm^{244}	1.4×10^{7}y	18.4y	7.6×10^{5}	2.6	1.0×10^{7}
Cf^{252}	66y	2.2y	30	3.5	2.6×10^{12}
Cf^{254}	60d	60d	~0		

*Alphas/fission is an inverse "figure of merit." A source with a low number of alphas/fission has relatively many fissions, and the neutron spectrum is less likely to be contaminated by (α,n) neutrons.

Table 19-3-4 summarizes the properties of the more important radionuclide neutron sources.

TABLE 19-3-4

COMMON NEUTRON SOURCES AND THEIR PROPERTIES

Source	Reaction	Half-life	Mean Neutron Energy (MeV)	Approx. Yield (n/sec·Ci)	Comments
^{124}SbBe	(γ,n)	60 days	0.024	1.6×10^{6}	Short half-life renders source unsuitable for many applications; has high-intensity gamma field.
^{210}PoB	(α,n)	138 days	2.3	6.0×10^{5}	
^{210}PoBe	(α,n)	138 days	4.5	2.5×10^{6}	
^{210}PoLi	(α,n)	138 days	0.5	6.1×10^{4}	Yield relatively low.
^{226}RaBe	(α,n) (γ,n)	1640 years 1640 years	3.9 0.3	1.5×10^{7} 1.2×10^{6}	High-intensity gamma field.
^{227}AcBe	(α,n)	21.8 years	4.6	~1.2×10^{6}	Not generally available.
^{228}ThBe	(α,n)	1.9 years	0.8	~2.5×10^{6}	Not generally available; high-intensity gamma field.
^{238}PuBe	(α,n)	87.8 years	4.5	1.8×10^{6}	Common; used for calibration of instruments.
^{239}PuBe	(α,n) (SF)+	24,360 years 24,360 years	4.5 1.2¶	1.5×10^{6} 0.5	Common; used for calibration of instruments.
^{239}PuF$_4$	(α,n)	24,360 years	0.7	~10^{6}	Physically relatively large, much self-absorption.
^{241}AmBe	(α,n) (SF)+	458 years	4.4¶ 1.2¶	2.2×10^{6} 1.9	Becoming common; good reactor starter source.
^{244}Pu	SF+	6600 years	1.2¶	3500*	Relatively weak for practical use.
^{244}Cm	SF+	18.4 years	1.2¶	1.5×10^{5}*	High gamma field.
^{252}Cf	SF+	2.65 years	1.2¶	4.3×10^{9}*	Coming into wide use as calibration and starter source.
^{249}Bk	SF+			3.6×10^{5}*	Not yet available in quantity.

+ SF = spontaneous fission.
¶ Mean energy of fission neutron spectrum.
* Yield is given in n/sec-g.

MACHINE SOURCES

Machine sources of neutrons include reactors and particle accelerators. In general, these sources are not widely employed because of their size. An exception is the so-called D-T (deuterium-tritium) generator, a small positive-ion accelerator that utilizes the D-T reaction for generating neutrons by bombarding a tritium-rich ($^{3}_{1}H$) target with positive deuterium ions ($^{2}_{1}H$):

$$^{2}_{1}H + {}^{3}_{1}H \longrightarrow {}^{4}_{2}He + {}^{1}_{0}n + Q \text{ (MeV)} \qquad \text{Eq. 19-3-3}$$

where:

Q = 17.6 MeV; all generated neutrons have the same energy, ~14 MeV; the other 3.6 MeV is associated with the liberated α particle.

Another neutron-generating machine sometimes used is a D-D (deuterium-deuterium) generator; it utilizes a D-D reaction. This machine is analogous to the D-T generator, except that the target is deuterium rich rather than tritium rich. Some commercial units are made with interchangeable targets in order that neutrons can be generated by either a D-D or a D-T reaction.

Accelerators of various kinds – betatrons, cyclotrons, linacs, etc. – can be used to produce neutrons from a wide variety of nuclear reactions. In addition to the D-D and D-T reactions, the (p,n) reaction is widely used in accelerators. In this reaction, a proton is accelerated into a target nucleus to produce neutrons:

$$ {}^{1}_{1}H + {}^{A}_{Z}X \longrightarrow {}^{A}_{(Z+1)}Y + {}^{1}_{0}n + Q \qquad \text{Eq. 19-3-4} $$

where:

X = target nucleus.

Y = product after reaction.

(In general, fission-produced neutrons from reactors are used for research, educational, and analytical purposes as well as for power production; the remaining modules in this course provide a more comprehensive consideration of this area.)

ENERGY OF NEUTRONS

Neutron sources can produce neutrons over a wide range of energies, ranging from about 10^{-4}eV to about 1000 MeV. Since the type of nuclear reaction or interaction with matter is strongly dependent upon neutron energy, neutrons often are classified according to their energy. There are four common categories (Table 19-3-5).

TABLE 19-3-5
CLASSIFICATION OF NEUTRONS

Type	Energy	Principal Interaction	Comments
Thermal	0.025 eV (at 22°C)	Capture	Mean velocity = 2200 m/sec Maxwellian distribution of energies.
Intermediate (Resonance Neutrons)	0.5 ev – 10 keV	Elastic scattering and capture	Absorption follows 1/v law and shows absorption peaks.
Fast	10 keV – 20 MeV	Elastic scattering	Most neutrons are ejected from nuclei with these energies.
Relativistic	>20 MeV	Spallation	Relativistic mass much greater than rest mass.

INTERACTION OF NEUTRONS WITH MATTER

Since the neutron has no electrical charge, it can easily interact with the nucleus of an atom. Interaction with an electron is rare; therefore, neutrons cause very little primary ionization.

Elastic scattering is the most likely type of interaction between neutrons and low-atomic-number absorbers. At each elastic collision, the neutron loses energy until it is reduced to thermal energy. Neutrons are slowed down by this process. Low-atomic-number materials, such as water and paraffin, often are used.

Inelastic scattering occurs when a neutron is captured by the nucleus and a gamma ray or a particle and neutron(s) are emitted. The neutron(s) that are emitted may or may not be the incident neutron. When the neutron is captured, the

nucleus reaches a higher energy state and returns to its ground state by emitting a gamma ray or particle. Capture reactions can occur with thermal, intermediate, or fast neutrons. The type of radiation emitted is a function of the energy of the incident neutron:

- Thermal neutrons cannot transfer enough energy to a nucleus to cause a particle to be emitted. Gamma rays are emitted instead. This process, described by Equation 19-3-5, is called a "(n,γ)" reaction, also known as "activation":

$$ {}^{A}_{Z}X + {}^{1}_{0}n \longrightarrow {}^{(A+1)}_{Z}X + \gamma \qquad \text{Eq. 19-3-5} $$

- Intermediate energy neutrons will transfer enough energy to the nucleus to allow a particle (proton, alpha) to be ejected. Thus, a (n,p) or (n,α) reaction will result; this reaction is called a "transmutation" since a new element is formed:

$$ (n,p): \quad {}^{A}_{Z}X + {}^{1}_{0}n \longrightarrow {}^{A}_{Z-1}X + {}^{1}_{1}p \qquad \text{Eq. 19-3-6} $$

$$ (n,\alpha): \quad {}^{A}_{Z}X + {}^{1}_{0}n \longrightarrow {}^{A-3}_{Z-2}X + {}^{4}_{2}He \qquad \text{Eq. 19-3-7} $$

- Fast neutrons contribute sufficient energy to the nucleus to make possible the emission of two particles. This is a (n, 2n) reaction:

$$ {}^{A}_{Z}X + {}^{1}_{0}n \longrightarrow {}^{A-1}_{Z}X + {}^{1}_{0}n + {}^{1}_{0}n \qquad \text{Eq. 19-3-8} $$

- Relativistic neutron reactions can cause even more particles to be emitted because of the high energy of the neutron; this is a (n, 2np) reaction:

$$ {}^{A}_{Z}X + {}^{1}_{0}n \longrightarrow {}^{A-2}_{Z-1}X + {}^{1}_{0}n + {}^{1}_{0}n + {}^{1}_{1}p \qquad \text{Eq. 19-3-9} $$

CROSS SECTIONS

A neutron cross section is expressed in units of area, usually cm^2. Because the effective area of a nucleus for a particular nuclear reaction is small, a new unit of area was created to eliminate the use of large negative exponents. This unit, the "barn," is equal to 10^{-24} cm^2/atom – approximately the cross-sectional area of a typical nucleus.

Many factors influence the value of a cross section:

- Type of reaction.
- Energy of incident neutron.
- Material undergoing reaction.

Cross section, commonly expressed in barns per atom or microscopic cross section, is symbolized by σ. There is a microscopic cross section for each particular type of reaction.

Total cross section, the sum of all other cross sections (e.g., scattering, capture, fission, etc.), can be expressed mathematically as follows:

$$\sigma_{tot} = \sigma_{scattering} + \sigma_{capture} + \sigma_{fission} + \cdots$$

or

$$\sigma_t = \sigma_s + \sigma_c + \sigma_f \qquad \text{Eq. 19-3-10}$$

Certain types of cross sections tend to predominate at specific energies, and may show recognizable trends as a function of energy. The greater the neutron velocity, the lower the capture cross section:

$$\sigma_c \propto \frac{1}{v} \qquad \text{Eq. 19-3-11}$$

where:

v = velocity of neutron.

There is a greater cross section with lower neutron velocity; consider the slower neutrons as spending more time in the vicinity of the nucleus and having proportionately more time to interact. Because the kinetic energy is given by $E = 1/2\ mv^2$ (and $v = \sqrt{2E/m}$), capture cross section is related to energy, as indicated by Equation 19-3-12.

$$\sigma_c \propto \frac{1}{v}$$

$$\sigma_c \propto \frac{1}{\sqrt{\frac{2E}{m}}} = \frac{1}{\sqrt{\frac{2}{m}} \times \sqrt{E}} \qquad \text{Eq. 19-3-12}$$

Since 2/m is a constant, it does not affect proportionality. And cross section is a function of energy:

$$\sigma_c \propto \frac{1}{\sqrt{E}} \qquad \text{Eq. 19-3-13}$$

Neutron cross sections often are plotted as a function of neutron energy for a particular nuclide. The plot usually is a log-log plot; that is, the log of the cross section (σ) is plotted against the log of the neutron energy (E). This arrangement permits several magnitudes of energies and cross sections to be depicted on a relatively small, smoothed, hypothetical plot (Figure 19-3-1). In general,

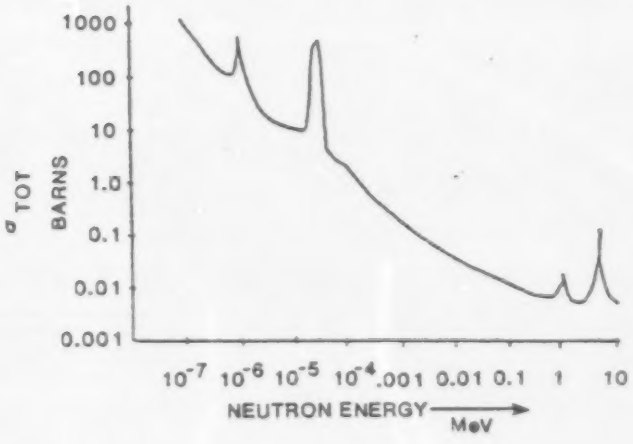

Figure 19-3-1. Neutron cross section versus neutron energy.

the 1/V relationship seems to be followed. In this particular example, irregularities appear as spikes at various points of the curve. Neutrons at these specific energies have a much greater chance of interacting than do neighboring neutrons of slightly greater or lesser energies. The

sharp increases in cross section are known as "resonances"; the prime reactions that occur at these specific points are "resonance reactions."

Resonances are a function of the energy levels within the nucleus. Some elements, such as boron, have few resonances and follow closely the 1/V relationship over a wide energy range. Others, such as cadmium and indium, have many resonances, their cross-section spectra are quite complex; for example, the capture cross section of cadmium shows a resonance for neutrons with energies of about 0.176 eV. (At 0.176 eV, the cross section is 7200 barns. At 1 eV, the cross section is only 20 barns; and at 10 eV, about 4 barns.) Indium produces 3 resonance peaks between 1 and 10 eV, and 8 peaks between 10 and 100 eV (Figure 19-3-2). The largest of the cross-section peaks, at 1.4 eV has a value of about 25,500 barns.

Indium and cadmium are good neutron absorbers, particularly for slow and thermal neutrons. Thus, they sometimes are used as thermal neutron shields or as poisons to absorb neutrons in a reactor. Boron is a good neutron absorber, but it has few resonances.

In general, capture cross sections predominate at low and intermediate energies, and particle reactions [(n,p), (n,α), etc.] tend to predominate at higher energies.

The value of the capture cross section is a function of neutron energy. Figure 19-3-3 plots capture cross section (probability of capture) versus neutron energy.

This graph demonstrates that as neutrons lose energy, the probability of an interaction with the nucleus increases. This is the basis for neutron shielding design: neutrons are slowed down and captured.

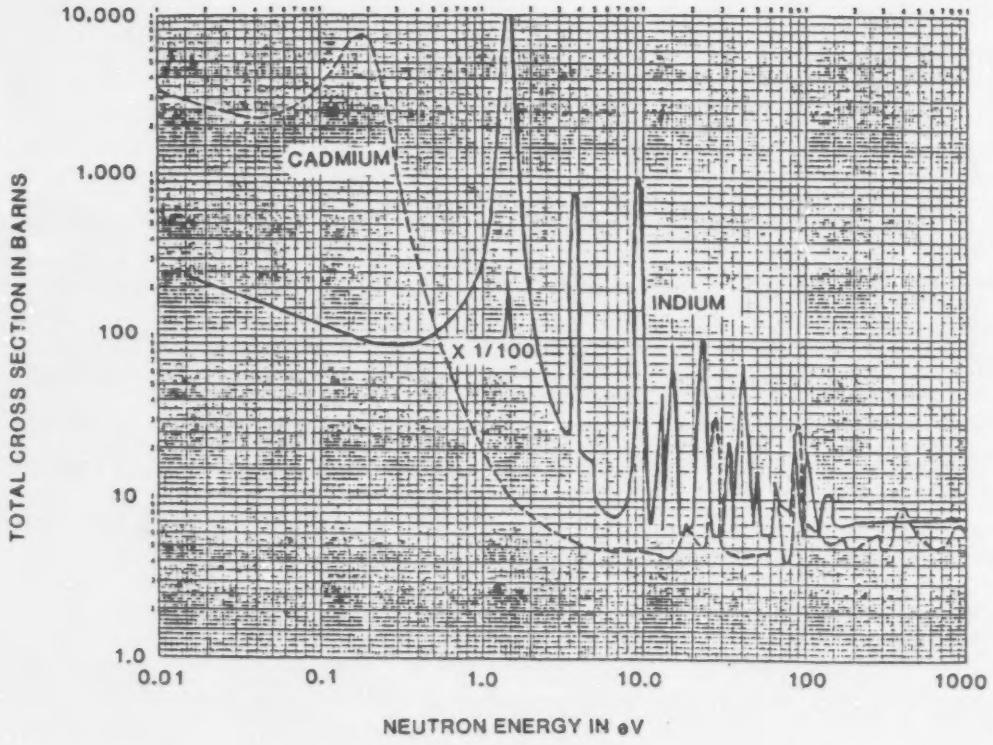

Figure 19-3-2. Typical neutron cross sections.

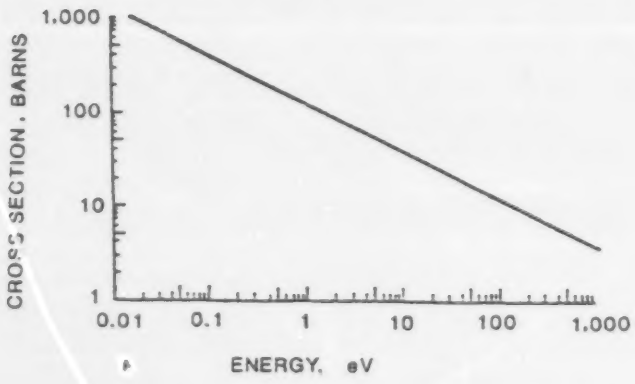

Figure 19-3-3. Neutron capture cross section for boron.

The slow neutron range often is referred to as the "1/V region" since the cross section is inversely proportional to the neutron velocity. Once the cross section (σ_1) at a specific energy (E_1) is known, the cross section (σ_2) for any other energy (E_2) within this range can be calculated from Equation 19-3-14.

$$\frac{\sigma_2}{\sigma_1} = \sqrt{\frac{E^1}{E^2}} \qquad \text{Eq. 19-3-14}$$

EXAMPLE

Given: The cross section for a boron reaction is 350 barns for 0.1-eV neutrons.

Find: The cross section for 100-eV neutrons.

Solution:

$$\sigma_2 = \frac{E^1}{E^2} \qquad \text{Eq. 19-3-15}$$

$$= 350 \text{ barns} \quad \frac{0.1}{100}$$

$$\sigma_2 = 11 \text{ barns.}$$

MACROSCOPIC CROSS SECTION

A macroscopic cross section, symbolized by Σ, is mathematically equal to the product of N and σ:

$$\Sigma = N\sigma \qquad \text{Eq. 19-3-16}$$

where:

N = number of absorber atoms/cm^3.

σ = microscopic cross section.

Thus, the macroscopic cross section has units of reciprocal (or inverse) length and is usually expressed as cm^{-1} or "per centimeter." It may be thought of as the probability of a neutron interaction per unit length (or cm) of a material. In addition, there are individual macroscopic cross sections for capture (Σ_c), scattering (Σ_s), and fission (Σ_f).

The cross section, σ (called "microscopic cross section" to distinguish it from "macroscopic cross section," Σ) refers to one atom. It is dependent only upon the material and the neutron energy. Macroscopic cross section, on the other hand, refers to the cross section of one cm^3 of atoms. It depends upon atomic density, on the different materials present, and on neutron energy.

An important application of macroscopic cross section is in the calculation of neutron shielding. Materials such as paraffin and polyethylene contain many elements; and, rather than being determined by multiple microscopic cross sections, a macroscopic cross section is employed in these calculations.

NEUTRON FLUX

"Neutron flux" is the number of neutrons passing through a unit area in unit time. If an absorber is placed before a neutron source, the change in neutron flux is exponential. This change in flux can be calculated from Equation 19-3-17.

$$I = I_o e^{-\sigma N t} \qquad \text{Eq. 19-3-17}$$

where:

I_o = the initial neutron flux in $n/cm^2/sec$.

I = the neutron flux at a point beyond the absorber in $n/cm^2/sec$.

σ = microscopic cross section in barns/atom. A barn is 10^{-24} cm^2.

N = the number of absorber atoms per cm^3.

t = the shield thickness in cm.

e = base of natural logarithms.

The number of absorber atoms per cm^3 (N) can be calculated by use of Equation 19-3-18.

$$N = \frac{\text{Avogadro's Number } (6.023 \times 10^{23} \text{ atoms/mole})}{\text{Atomic Weight of Absorber}} \times \text{Density of Absorber} \qquad \text{Eq. 19-3-18}$$

EXAMPLE

Given: A 10-cm thick boron absorber is placed before a beam of 10-MeV neutrons with a flux, I_o of 10^5 $n/cm^2/sec$. σ of boron = 1.4 barns.

Find: The change in flux.

Solution:

Use $I = I_o e^{-\sigma n t}$

where:

$I_o = 10^5$ $n/cm^2/sec$.

σ = 1.4 barns = 1.4 x 10^{-24} cm^2.

$$N = \left(\frac{6.023 \times 10^{23} \text{ atoms/mole}}{10.8 \text{ g/mole}}\right)(2.34 \text{ g/cm}^3).$$

where:

6.023 x 10^{23} atoms/mole = Avogadro's number.

10.8 g/mole = atomic weight of boron.

2.34 g/cm^3 = density of boron.

N = 1.3 x 10^{23} $atoms/cm^3$.

t = 10 cm.

$$I = I_o e^{-\sigma n t}$$

$$= \left(10^5 \frac{N}{cm^2 \cdot sec}\right) e^{-(1.4 \times 10^{-24} \text{ cm}^2)(1.3 \times 10^{23} \text{ atoms/cm}^3)(10 \text{ cm})}$$

$$= \left(10^5 \frac{N}{cm^2/sec}\right) e^{-1.82}$$

$$= \left(10^5 \frac{N}{cm^2/sec}\right)(0.16)$$

$$I = 1.62 \times 10^4 \frac{N}{cm^2/sec}.$$

NEUTRON SHIELDING

Neutron shielding is based upon decreasing the energy of fast neutrons by elastic scattering and then absorbing the resultant low-energy neutrons. Low-atomic-number materials such as hydrogen, boron, and paraffin are used for shielding materials. Since the maximum transfer of kinetic energy from the neutron to the nucleus occurs when the mass of the nucleus equals the mass of the neutron, these materials are particularly suitable.

When a fast neutron is slowed down by collisions with nuclei, it becomes a thermal neutron. The thermal neutron eventually is absorbed by the shielding material. The distance that the fast neutron travels during this process is the "fast diffusion length." The distance that a thermal neutron travels before absorption is the "thermal diffusion length." Thus, the thermal neutron length may be defined as "the thickness of an absorbing medium that attenuates a beam of thermal neutrons by a factor of e" (the base of natural logarithms). This factor is employed since the change in neutron flux is exponential.

For material of thickness t(cm) with thermal diffusion length L(cm), the attenuation of the thermal neutrons is given by Equation 19-3-19.

$$n = n_o e^{-t/L} \qquad \text{Eq. 19-3-19}$$

where:

n_o = initial neutron flux in $n/cm^2/sec$.
n = neutron flux after passing through the absorber.
e = base of natural logarithms.
t = thickness of absorbing material in cm.
L = thermal diffusion length in cm.

This equation applies to all materials with low absorption cross section (water, carbon). For materials with high absorption cross sections, such as boron or cadmium, Equation 19-3-17 can be used.

Values for fast and thermal diffusion lengths can be obtained from tables such as Table 19-3-6.

TABLE 19-3-6

VALUES OF DIFFUSION LENGTHS OF SOME MATERIALS

Material	Fast Diffusion Length	Thermal Diffusion Length
H_2O	5.75 cm	2.88 cm
D_2O	11	171
Be	9.9	25
Carbon (graphite)	17.3	50

Shielding normally is arranged in such a manner that a specified dose equivalent (DE) is obtained. The dose equivalent from neutrons is a function of neutron energy. To find the flux in $n/cm^2/sec$ necessary to achieve a specified dose rate, the technician can consult published data. Table 19-3-7 lists the neutron flux corresponding to a DE of 100 millirems per week.

To calculate shielding requirements necessary to achieve a specific DE, the technician obtains values for neutron energy and initial flux, obtains the flux corresponding to the desired DE from tables like Table 19-3-7, and uses either Equation 19-3-17 or Equation 19-3-18 to calculate shielding thickness.

TABLE 19-3-7

NEUTRON QUALITY FACTOR AND FLUX CORRESPONDING TO 100 MILLIREMS PER WEEK

Neutron Energy MeV	QF	$n/cm^2/sec$ 100 rem/week
Thermal	3	670
0.0001	2	500
0.005	2.5	570
0.02	5	280
0.1–	8–	80–
0.5	10	30
1.0	10.5	18
2.5	8	20
5.0	7	18
7.5	7	17
10.0	6.5	17
10–30		10

EXAMPLE

Given: Neutron energy is 100 keV. Initial flux is 10^5 $n/cm^2/sec$. Boron will be used as shielding material.

Find: The thickness of the boron shield that will reduce the flux to a level corresponding to 100 mrem per week.

Solution: Consult Table 19-3-7 to obtain flux, corresponding to desired dose rate. 100-keV neutrons with

a flux of 80 n/cm²/sec will yield a DE of 100-mrem/wk. Use Equation 19-3-17 to find shielding thickness. This equation is used because boron has a high absorption cross section.

$$I = I_o e^{-\sigma N t} \qquad \text{(Eq. 19-3-17)}$$

where:

I_o = the initial neutron flux: 10^5 n/cm²/sec.

I = the neutron flux corresponding to the desired dose rate: 80 n/cm²/sec.

e = the base of natural logarithms.

σ = microscopic cross section of boron in barns: 4.2 barns = 4.2×10^{-24} cm².

N = the number of absorber atoms per cm³: 1.3×10^{23} atoms/cm³ (calculated in previous example).

Solve for t:

$$e^{\sigma N t} = \frac{I_o}{I}$$

$$\ln(e^{\sigma N t}) = \ln\left(\frac{I_o}{I}\right)$$

$$\sigma N t = \ln\left(\frac{I_o}{I}\right).$$

$$t = \frac{1}{\sigma N} \ln\left(\frac{I_o}{I}\right)$$

$$= \frac{1}{(4.2 \times 10^{-24}\ \text{cm}^2)(1.3 \times 10^{23}\ \text{atoms/cm}^3)} \ln\left(\frac{10^5}{80}\right)$$

$$= \frac{1}{0.55}\ (7.13)$$

$$t = 13.1\ \text{cm}.$$

Boron 13.1 cm thick is required if flux is to be reduced to a level corresponding to 100 mrem per week.

In the example above, the desired neutron flux was obtained from tables. Occasionally, the neutron flux must be calculated after neutrons have passed through shielding for a specific source and geometry without the aid of these tables. Methods of calculation vary with the geometry of the source and the shielding. Calculation of neutron flux for a point source of thermal neutrons surrounded by a spherical shield is a typical procedure.

For a point source of thermal neutrons surrounded by a spherical shield of radius R and thermal diffusion length L, the neutron flux at the outer surface of the sphere is given by Equation 19-3-20.

$$\phi = \frac{N_o}{4\pi RL \sinh (R/L)} \qquad \text{Eq. 19-3-20}$$

where:

N_o = initial neutron intensity (n/sec).
R = radius of shield.
L = thermal diffusion length of shield.
sinh (R/L) = hyperbolic sin of (R/L).
ϕ = neutron flux at outer surface of sphere.

This equation is used if L>R. If L<R then Equation 19-3-21 is used.

$$\phi = \frac{N_o}{2\pi RL} e^{-R/L} \qquad \text{Eq. 19-3-21}$$

where:

e = base of natural logarithms.

EXAMPLE

Given: A spherical carbon shield 40 cm in diameter surrounding a point source of thermal neutrons whose initial intensity is 10^7n/sec.

Find: The flux of thermal neutrons at the outer surface of the shield.

Solution: R = 20 cm (½ 40 cm)

L = 50 cm (obtained from Table 19-3-6). Since L>R, use Equation 19-3-20:

$$\phi = \frac{N_o}{4\pi RL \sinh (R/L)}$$

$$\phi = \frac{10^7 \text{ n/sec}}{4\pi \text{ (20 cm)(50 cm) } \sinh (\frac{20}{50})}$$

where:

$$\sinh (\frac{20}{50}) = 0.41.$$

$$\phi = \frac{10^7 \text{ n/sec}}{4\pi \text{ (}10^3 \text{ cm}^2\text{)(0.41)}}$$

$$\phi = 1941 \text{ n/cm}^2\text{/sec.}$$

Neutron flux at the surface of the sphere is 1941 n/cm²/sec.

The example below illustrates procedures used when L<R.

EXAMPLE

Given: A spherical carbon shield 200 cm in diameter surrounding a point source of thermal neutrons whose initial flux is 10^7 n/sec.

Find: The flux of thermal neutrons at the outer surface of the shield.

Solution: R = 100 cm (½ 200 cm).

L = 50 cm (from Table 19-3-6). Since L<R use Equation 19-3-21.

$$\phi = \frac{N_o}{2\pi\ RL}\ e^{-R/L}$$

$$= \frac{10^7\ \text{n/sec}}{2\pi\ (10^2\ \text{cm})(50\ \text{cm})}\ e^{-100/50}$$

$$= \frac{10^7\ \text{n/sec}}{3.14 \times 10^4\ \text{cm}}\ (0.135)$$

$$\phi = 43\ \text{n/cm}^2\text{/sec}.$$

Neutron flux at the surface of the sphere is 43 $n/cm^2/sec$.

EXTERNAL NEUTRON DOSE

External dose from fast neutrons can be calculated from Equation 19-3-22.

$$D = \frac{\phi E\ [N_i\ \sigma_i\ f_i]}{100\ \text{ergs/g rad}} \qquad \text{Eq. 19-3-22}$$

where:

ϕ = neutron flux in $n/cm^2/sec$.

E = neutron energy in ergs.

N_i = number of atoms per gram of element i.

σ_i = scattering cross section of element i in cm^2 for neutrons of energy E.

f_i = mean fractional energy transferred to scattered atom of a specific mass during collision with a neutron.

Table 19-3-8 gives the number of atoms per gram (N) and mean-fractional energy transfer of four elements found in soft tissue.

TABLE 19-3-8
VALUES OF N AND f FOR ELEMENTS FOUND IN SOFT TISSUE

Element	N, atoms/g	f
H	6.70×10^{22}	0.500
C	4.18×10^{21}	0.142
N	1.17×10^{21}	0.124
O	2.93×10^{22}	0.111

Dose rate is determined from Equation 19-3-22. Neutron flux and values of n, f, and σ must be determined for each element. The equation then is solved for D.

EXAMPLE

Given: 5-MeV neutrons with a flux of 100 $n/cm^2/sec$ are incident of soft tissue. Values for N and f are stated in Table 19-3-8. Values for σ are –

Element: H – 1.50×10^{-24} cm^2.

C – 1.65×10^{-24} cm^2.

N – 1.00×10^{-24} cm^2.

O – 1.55×10^{-24} cm^2.

Find: Dose rate to the tissue.

Solution: Use Equation 19-3-22:

$$D = \frac{\phi E \left[N_i \sigma_i f_i\right]}{100 \text{ ergs/g rad}}$$

Substitute for i:

$$D = \phi E[N_o\sigma_o f_o + N_c\sigma_c f_c + N_n\sigma_n f_n + N_H\sigma_H f_H]$$

where:

σ = 100 $n/cm^2/sec$.

E = 5-MeV neutrons. Convert to ergs:

$$\left(\frac{5 \text{ MeV}}{\text{neutrons}}\right)\left(\frac{1.6 \times 10^{-6} \text{ ergs}}{\text{MeV}}\right)$$

E = 8×10^{-6} ergs/neutron.

N_o = 2.93×10^{22} atoms/g.

σ_o = 1.55×10^{-24} cm^2.

f_o = 0.111.

N_c = 4.18×10^{21} atoms/g.

σ_c = 1.65×10^{-24} cm^2.

f_c = 0.142.

N_n = 1.17×10^{21} atoms/g.

σ_n = 1.00×10^{-24} cm^2.

f_n = 0.124.

N_H = 6.70×10^{22} atoms/g.

$\delta_H = 1.50 \times 10^{-24}$ cm^2.

$f_H = 0.500$.

Solve for D:

$$D = \frac{(10^2 \text{ n/cm}^2\text{/sec})(8 \times 10^{-6} \text{ erg/n})}{(10^2 \text{ erg/g·rad})} [2.93 \times 10^{22} \text{ atoms/g}) (1.55 \times 10^{-24} \text{ cm}^2) (0.111)$$
$$+ (4.18 \times 10^{21})(1.65 \times 10^{-24})(0.142)$$
$$+ (1.17 \times 10^{21})(1.00 \times 10^{-24})(0.124)$$
$$+ (6.70 \times 10^{22})(1.50 \times 10^{-24})(0.500)]$$
$$= (8 \times 10^{-6} \frac{\text{rad}}{\text{sec}})(5.64 \times 10^{-2})$$
$$= (4.51 \times 10^{-7} \frac{\text{rad}}{\text{sec}}) (\frac{10^3 \text{ mrad}}{\text{rad}}) (\frac{3.6 \times 10^3 \text{ sec}}{\text{hr}})$$

D = 1.62 mrad/hour.

The dose rate to tissue is 1.62 mrad/hour.

EXERCISES

1. Calculate the change in flux when a beam of 5-MeV neutrons of initial flux 10^5 n/cm^2/sec passes through a 7-cm thick boron absorber.
2. Calculate the dose rate to soft tissue for 15-Mev neutrons with a flux of 500 n/cm^2/sec.
3. Calculate the thickness of a boron shield that will reduce a beam of 7.5-MeV neutrons with initial flux of 10^5 n/cm^2/sec to a flux level corresponding to 100 mrem/wk.
4. A spherical carbon shield 30 cm in diameter surrounds a point source of thermal neutrons whose initial inten-

sity is 10^6 n/sec. Calculate the flux of thermal neutrons at the outer surface of the shield.

5. The cross section for a boron reaction is 350 barns for 0.1-eV neutrons. Calculate the cross section for 50-eV neutrons.

MATERIALS

Neutron source, Pu-Be, 80 g (~5 Ci) with intensity of 8.7×10^6 n/sec.

Gamma-ray source, ^{60}Co (~15 mCi).

Neutron survey instrument, Bonner Spheres, and Nemo (Texas Nuclear). Five or six Bonner spheres with diameters of 2, 3, 5, 8, 10, and 12 inches should be available.

Meter stick.

PROCEDURES

1. Measure the gamma radiation response with the Bonner Spheres and the ^{60}Co source at approximately 100 cm and 200 cm, and complete Data Table 19-3-1.
2. Determine instrument response versus distance. Start at 50 cm from the source, and complete Data Table 19-3-2.
3. At 200-cm distance, measure the response of each of the Bonner Spheres, and complete Data Table 19-3-3.

4. Check the directional dependence of a Bonner Sphere by rotating it in the neutron field and observing response. Complete Data Table 19-3-4.

Questions:

1. Does the instrument respond to gamma radiation? Since gamma radiation is present in many neutron fields, will instrument response to gamma radiation produce inaccurate neutron measurements?
2. For Step 2 above, determine the neutron flux at various distances, and then graph this flux ($n/cm^2/sec$) versus activity (cpm) measured. Does the graph produce a straight line? (See Figure 19-3-4.)

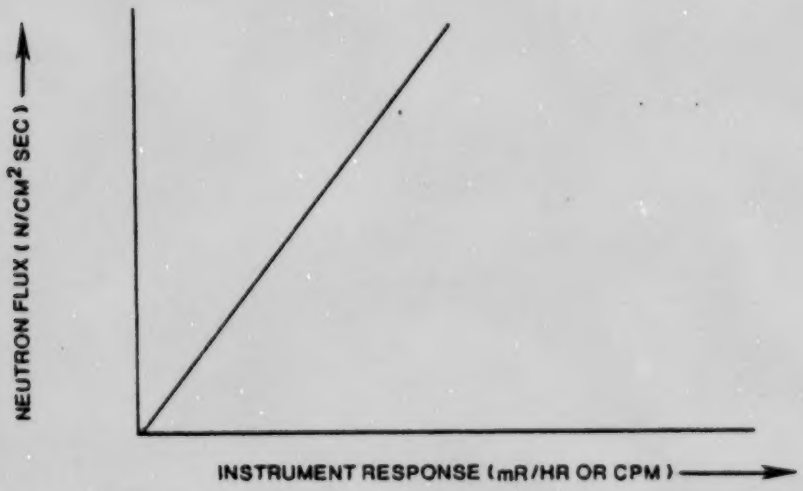

Figure 19-3-4. Flux versus activity curve.

3. Explain how response varies with sphere diameter.
4. Describe any directional dependence of the Bonner Sphere.

DATA TABLE 19-3-1

Distance (cm)	Expected Exposure Rate (mR/hour)	Measured Exposure Rate (mR/hour)
100 cm		
200 cm		

DATA TABLE 19-3-2

Distance (cm)	Response (mR/hour)	Expected Flux ($n/cm^2/sec$)(Calc)
50		
75		
100		
125		
150		
175		
200		
225		
250		

DATA TABLE 19-3-3

Sphere Diameter	Response (mR/hour) at 200 cm
2	
3	
5	
8	
10	
12	

DATA TABLE 19-3-4

Rotation of Sphere	Response mR/hour
0°	
90°	
180°	
270°	

REFERENCE MATERIALS

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1. List three radionuclide neutron sources, and describe their characteristics.

2. List and describe the energy classifications of neutrons.

3. Describe neutron interaction with the nucleus.

4. Describe the relationship of absorption cross section and neutron energy.

5. Calculate the change in flux when a beam of 1-MeV neutrons of initial flux 10^6 n/cm^2/sec passes through a boron absorber of thickness 5 cm.

6. Find the dose equivalent to tissue for 5-MeV neutrons with a flux of 250 n/cm^2/sec.

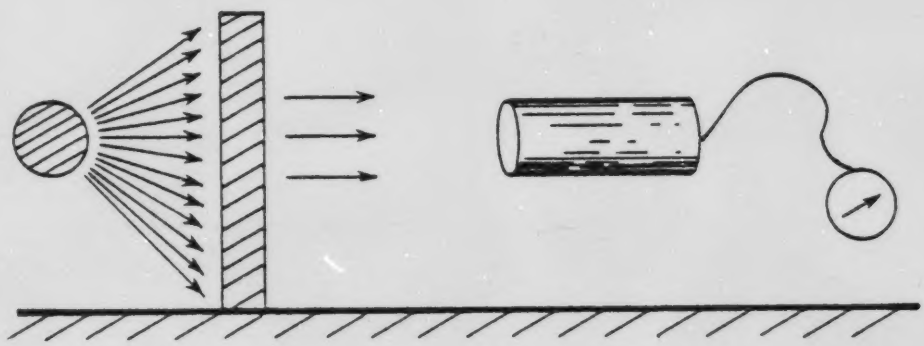
7. The cross section for a boron reaction is 350 barns for 0.1-eV neutrons. Calculate the cross section for 0.60-eV neutrons.

8. Calculate the thickness of a boron shield that will reduce a beam of 5-MeV neutrons having an initial flux of 10^6 n/cm^2/sec to a level corresponding to 100 mrem/wk.

NUCLEAR TECHNOLOGY

MODULE 19 - 4

GAMMA ATTENUATION



RADIATION SHIELDING

COURSE 19

CENTER FOR OCCUPATIONAL RESEARCH AND DEVELOPMENT

COURSE MODULES

COURSE 19 RADIATION SHIELDING

MODULE 19-1 General Effect of Radiation Attenuation
MODULE 19-2 Beta and Alpha Particle Attenuation
MODULE 19-3 Neutron Attenuation
MODULE 19-4 Gamma Attenuation
MODULE 19-5 Background Radiation Reduction Techniques
MODULE 19-6 Backscattering and Wall Scattering of Radiation

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INTRODUCTION

Gamma radiation consists of high-energy electromagnetic rays. These rays are highly penetrating and still may be detected after passing through several inches of steel; therefore gamma radiation presents a more significant hazard from external radiation exposure than from any other form of radiation. Shielding is an important method of protection from its effects.

Correct determination of shielding requirements demands a thorough understanding of gamma sources and exposure rates. This module discusses exposure rates from different gamma sources; demonstrates the calculation of these exposure rates; and explores interactions of gamma rays with matter. It also introduces the concept of "build-up factor" and explains how this factor is used to calculate gamma shielding.

MODULE PREREQUISITES

The student should have completed a course in basic algebra; Course 1, "Radiation Physics"; and Course 2, "Radiation Protection II."

OBJECTIVES

Upon completion of this module, the student should be able to:

1. Define these terms:
   a. Rayleigh Scattering.
   b. Thompson Scattering.
   c. Mössbauer Resonance Scattering.
   d. Photodisintegration.
   e. Photoelectric Effect.
   f. Compton Effect.
   g. Pair Production.
   h. Build-up Factor.
2. Given the gamma-ray constant, the distance from source, and the source activity in curies, calculate the exposure rate for a specified radionuclide.
3. Given the activity and the length of a source, distance from end of source to point of interest, and perpendicular distance from source to point of interest, calculate the exposure rate due to gamma radiation at the point of interest for line, ring, and disc sources.
4. Describe how "poor geometry" affects the readings of a radiation-detection instrument.
5. Given the quantity and type of a gamma source, the gamma-ray constant, the build-up factor, and the desired exposure rate at a specified distance from the source, calculate shielding requirements.

SOURCES OF ELECTROMAGNETIC RADIATION

Two kinds of electromagnetic radiation are emitted by radioactive elements – X-radiation (X-rays) and gamma radiation. X-rays are created in reactions outside the nucleus; gamma rays are created within the nucleus.

X-rays are produced when a stream of fast-moving electrons is slowed down upon striking a suitable target. Collisions between these electrons and target atoms cause electron transitions within atomic orbital electron shells, creating photons of energy called "characteristic X-rays."

Gamma rays are created when the nucleus of a radioactive atom emits an alpha or beta particle and the daughter nucleus is left in a high-energy state. The excess energy usually is emitted as gamma radiation, bringing the nucleus to a more stable energy condition. Thus, gamma rays originate in the nuclei of atoms.

GAMMA RADIATION: INTERACTIONS WITH MATTER

Gamma radiation can interact with nuclei, with fields around nuclei, or with orbital electrons. These interactions can be elastic on inelastic, as discussed later in this module. At times, gamma rays are absorbed completely.

RAYLEIGH SCATTERING

Rayleigh Scattering is an interaction between a gamma ray and an orbital electron. In this interaction, the gamma ray collides with an electron and "bounces" off. The energy of the gamma ray is unchanged. This interaction involves gamma rays that lack the energy needed to cause excitation or ionization (Figure 19-4-1).

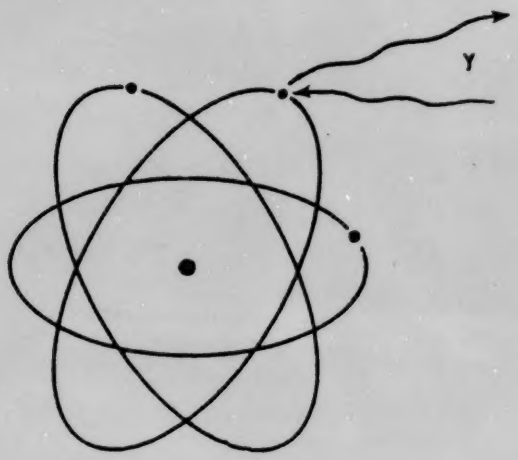


Figure 19-4-1.
Rayleigh Scattering.

THOMPSON SCATTERING

Thompson Scattering is another type of interaction between gamma rays and electrons (Figure 19-4-2).

The electron absorbs the gamma ray, oscillates in an excited state, and then radiates a gamma ray in a random direction. The collision is elastic.

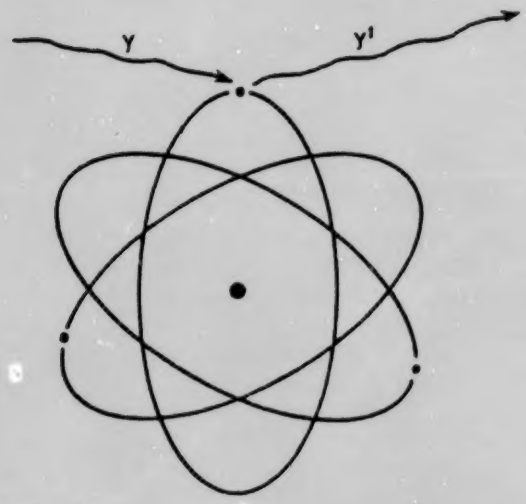


Figure 19-4-2.
Thompson Scattering.

MÖSSBAUER EFFECT

Gamma rays often interact with the nucleus. In the Mössbauer Effect, sometimes called "Nuclear Mössbauer Resonance Scattering" (Figure 19-4-3), the gamma ray is absorbed by the nucleus. The nucleus is excited momentarily and radiates another gamma ray. This collision is inelastic.

NUCLEAR TRANSFORMATION

A nuclear transformation or photodisintegration sometimes occurs. In this interaction, the gamma ray is completely absorbed by the nucleus (Figure 19-4-4). This absorption happens only when gamma rays that possess energies over 6 MeV strike the nucleus. When the gamma ray is absorbed, a particle, usually a neutron, is emitted.

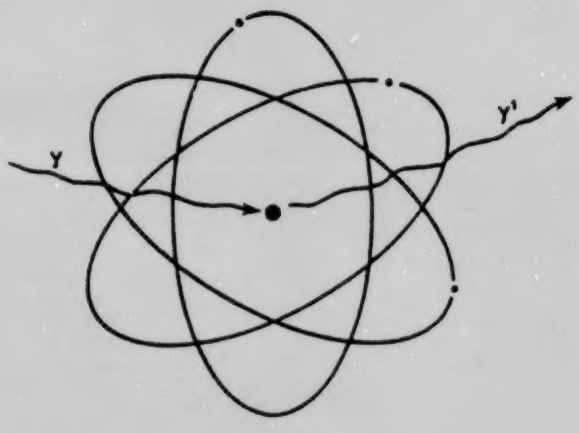


Figure 19-4-3.
The Mössbauer Effect.

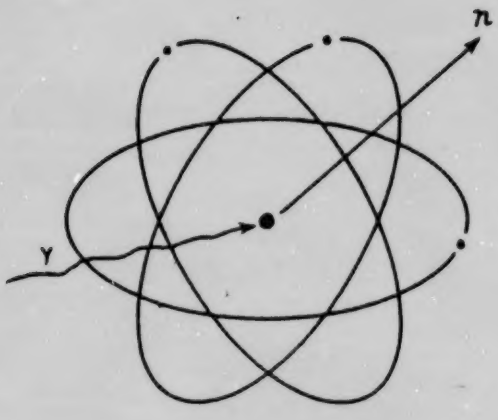


Figure 19-4-4.
Nuclear transformation.

A gamma photon can be absorbed in three different ways: (1) the photoelectric effect, (2) Compton Scattering, and (3) pair production. The probability of occurrence of the first two of these reactions increases with the number of electrons present, although pair production is most likely to occur near a heavy nucleus. Heavy materials such as lead and steel, therefore, are good gamma-ray absorbers.

A larger volume of a lighter material, that contains about the same number of electrons, is approximately as effective a shield as a small volume of heavy material. Regardless of the material used, the shield must reduce the intensity of gamma radiation from both the primary and the secondary origins. (Gamma rays from the reactor are primary; those from the shield itself, secondary.)

PHOTOELECTRIC EFFECT

The Photoelectric Effect (Figure 19-4-5) occurs when gamma rays that have energies below 0.5 MeV interact with orbital electrons. The gamma-ray energy is absorbed by the electron; and the electron is knocked out of its orbit, and an ion pair is formed. This effect often occurs in absorbers that have high atomic (Z) numbers. Lead and other high-Z materials are good absorbers of gamma rays. This interaction is inelastic.

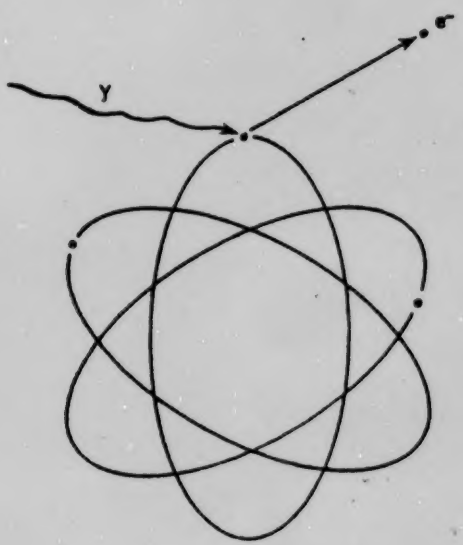


Figure 19-4-5.
The Photoelectric Effect.

COMPTON EFFECT

The Compton Effect is similar to the Photoelectric Effect; however, in the Compton Effect, the collision with the electron is elastic, and only a portion of the gamma-ray energy is given to the orbital electron (Figure 19-4-6).

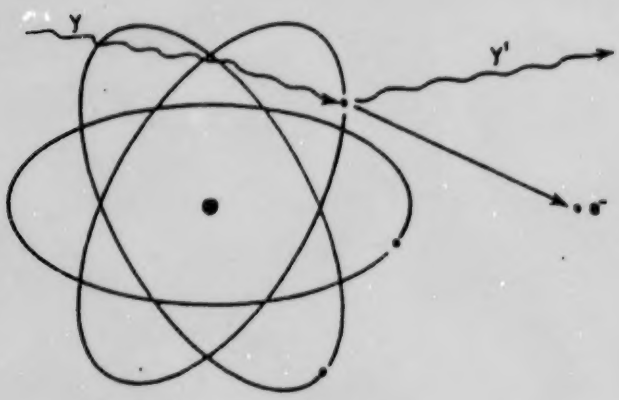


Figure 19-4-6.
Compton Effect.

The electron is ejected from its orbit, and the gamma ray is scattered in a different direction.

This effect occurs when gamma rays of medium energy (between 0.5 and 1.5 MeV) encounter absorbers of low-to-medium atomic number.

PAIR PRODUCTION

When gamma-ray energies are higher than 1.02 MeV, gamma rays can interact with the electromagnetic field of the nucleus. When this occurs, a pair of particles – an electron and a positron – are created. This process is called "Pair Production" (Figure 19-4-7).

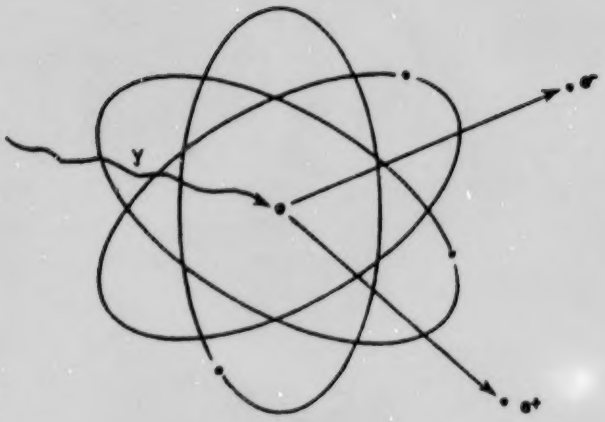


Figure 19-4-7.
Pair Production.

When the gamma ray encounters the electromagnetic field of the nucleus, gamma-ray energy is converted to an electron and a positron. Since these particles have a mass of 0.51 MeV each, gamma-ray energy must be at least 1.02 MeV (the energy equivalent to the rest mass of one electron and one positron). Any additional gamma-ray energy in excess of 1.02 MeV is given to the electron/positron pair as kinetic energy. The higher the atomic number and gamma-ray energy, the greater the opportunity for pair production.

Almost as soon as the positron is emitted, it is annihilated, producing two 0.511-MeV gamma rays. The electron gradually loses its energy by excitation, ionization, and bremsstrahlung.

EXTERNAL GAMMA DOSE

The external gamma exposure rate must be considered when setting up shielding. This rate is a function of the absorbed energy per unit mass of air at a specified distance from a point source. This relationship is expressed by

Equation 19-4-1.

$$ER = \Gamma \frac{A}{d^2}$$ Equation 19-4-1 (Equation = Eq.)

where

ER = exposure rate.

Γ = Specific Gamma-ray Constant (defined in next paragraph).

d = distance from the source.

A = source activity.

For a nuclide that emits gamma radiation, the Specific Gamma-ray Constant is the product of exposure rate at a given distance from a point source of that nuclide and the square of that distance divided by the activity of the source, neglecting attenuation. Published values of specific gamma-ray constants (Γ) are available in units of $\frac{R}{hr} \times \frac{m^2}{hr} \times \frac{cm^2}{mCi}$. Table 19-4-1 illustrates gamma radiation levels for one curie of some radionuclides. Once the Gamma-ray Constant for any nuclide is known, the constant can be used to calculate the external gamma dose for any activity distance. (Refer to the next example problem.)

TABLE 19-4-1

TABLE OF SPECIFIC GAMMA-RAY CONSTANTS

Nuclide	Γ	Nuclide	Γ	Nuclide	Γ
Actinium-227	-2.2	Gold-198	2.3	Potassium-43	5.6
Antimony-122	2.4	Gold-199	-0.9	Radium-226	8.25
Antimony-124	9.8	Hafnium-175	-2.1	Radium-228	-5.1
Antimony-125	-2.7	Hafnium-181	-3.1	Rhenium-186	-0.2
Arsenic-72	10.1	Indium-114m	-0.2	Rubidium-86	0.5
Arsenic-74	4.4	Iodine-124	7.2	Ruthenium-106	1.7
Arsenic-76	2.4	Iodine-125	-0.7	Scandium-46	10.9
Barium-131	-3.0	Iodine-126	2.5	Scandium-47	0.56
Barium-133	-2.4	Iodine-130	12.2	Selenium-75	2.0
Barium-140	12.4	Iodine-131	2.2	Silver-110m	14.3
Beryllium-7	-0.3	Iodine-132	11.8	Siver-111	-0.2
Bromine-82	14.6	Iridium-192	4.8	Sodium-22	12.0
Cadmium-115m	-0.2	Iridium-194	1.5	Sodium-24	18.4
Calcium-47	5.7	Iron-59	6.4	Strontium-85	3.0
Carbon-11	5.9	Krypton-85	-0.04	Tantalum-182	6.8
Cerium-141	0.35	Lanthanum-140	11.3	Tellurium-121	3.3
Cerium-144	-0.4	Lutecium-177	0.09	Tellurium-132	2.2
Cesium-134	8.7	Magnesium-28	15.7	Thulium-170	0.025
Cesium-137	3.3	Manganese-52	18.6	Tin-113	-1.7
Chlorine-38	8.8	Manganese-54	4.7	Tungsten-185	-0.5
Chromium-51	0.16	Manganese-56	8.3	Tungsten-187	3.0
Cobalt-56	17.6	Mercury-197	-0.4	Uranium-234	-0.1
Cobalt-57	0.9	Mercury-203	1.3	Vanadium-48	15.6
Cobalt-58	5.5	Molybdenum-99	-1.8	Xenon-133	0.1
Cobalt-60	13.2	Neodymium-147	0.8	Ytterbium-175	0.4
Copper-64	1.2	Nickel-65	-3.1	Yttrium-88	14.1
Europium-152	5.8	Niobium-95	4.2	Yttrium-91	0.01
Europium-154	-6.2	Osmium-191	-0.6	Zinc-65	2.7
Europium-155	-0.3	Palladium-109	0.03	Zirconium-95	4.1
Gallium-67	-1.1	Platinum-197	-0.5		
Gallium-72	11.6	Potassium-42	1.4		

EXAMPLE

Given: 100 mCi of ^{192}Ir (approximates a point source).

Find: The exposure rate at 10 feet.

Solution: Use Equation 19-4-1: $ER = \Gamma \frac{A}{d^2}$

Where $\Gamma = 4.8 \frac{R}{hr} \frac{cm^2}{mCi}$.

$A = 100$ mCi.

$d = 10$ feet $= 10$ ft $(\frac{30.48 \text{ cm}}{\text{ft}}) = 304.8$ cm.

Thus, $ER = (4.8 \frac{R}{hr} \frac{cm}{mCi}) \frac{(100 \text{ mCi})}{(304.8 \text{ cm})^2}$

$= 0.0052 \frac{R}{hr}$

$= (0.0052 \frac{R}{hr}) (\frac{10^3 \text{ mR}}{R})$

$= (0.0052) \quad (10^3 \frac{mR}{hr})$

$ER = 5.2 \frac{mR}{hr}$.

GEOMETRY OF SOURCES

Equation 19-4-1 can be used to calculate the gamma exposure rate for point sources only. It must be modified to give accurate results for sources of different geometries.

Line sources (line-shaped sources), such as pipes and needles, often are encountered (Figures 19-4-8 and 19-4-9).

The exposure rate for line sources is given by Equation 19-4-2.

$$ER = \frac{\Gamma A}{ah} [\tan^{-1} (\frac{a + b}{h}) - \tan^{-1} (\frac{b}{h})] \frac{R}{hr} \qquad \text{Eq. 19-4-2}$$

where

ER = exposure rate in $\frac{R}{hr}$ due to gamma radiation at the point of interest (P).

A = activity of the source in mCi.

a = length of the source in cm.

h = perpendicular distance from centerline of source to point of interest in cm.

b = distance from end of the source to the point of interest (P) in cm.

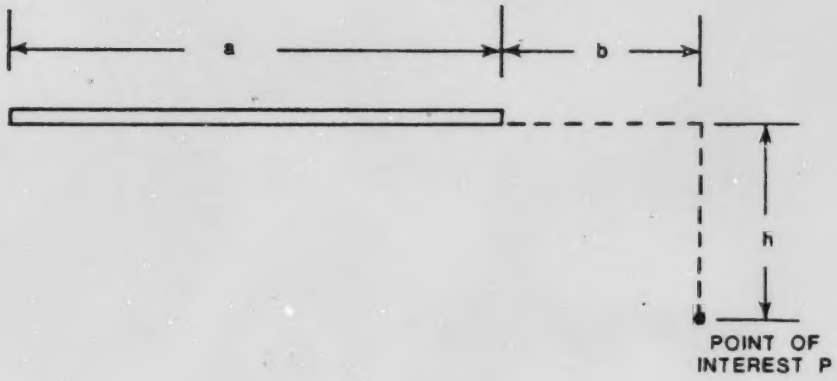


Figure 19-4-8. Line source.

The distance b can be negative, as illustrated in Figure 19-4-9.

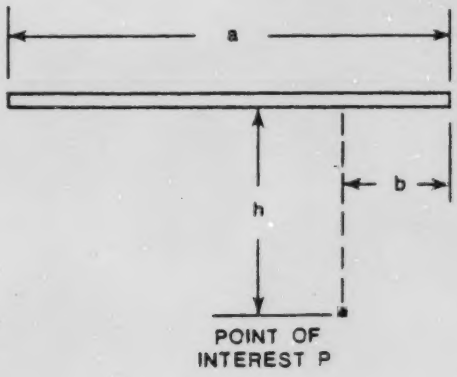


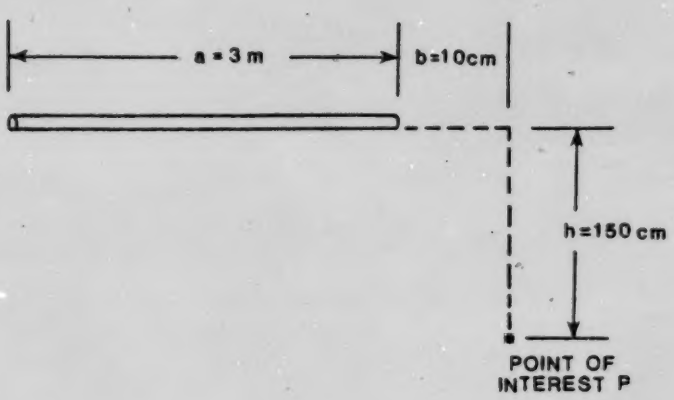
Figure 19-4-9. Line source with negative value "b."

The use of Equation 19-4-2 is illustrated in the following example.

EXAMPLE

Given: 10 Ci of ^{137}Cs distributed evenly along a pipe 3 m in length.

Find: The exposure rate ER at P, where b = 10 cm, and h = 150 cm.



Solution: Use Equation 19-4-2:

$$ER = \frac{\Gamma A}{ah}\left[\tan^{-1}\left(\frac{a+b}{h}\right) - \tan^{-1}\left(\frac{b}{h}\right)\right]\frac{R}{hr}$$

where $\Gamma = 3.3 \frac{R}{hr}\frac{cm^2}{mCi}$, (from Table 19-4-1)

$A = 10\ Ci = 10 \times 10^3\ mCi = 10^4\ mCi$,

$a = 3m$

$a = 300\ cm$

$b = 10\ cm$,

$h = 150\ cm$.

$$ER = \frac{(3.3 \frac{R}{hr} \frac{cm^2}{mCi})(10^4 \text{ mCi})}{(3 \times 10^2 \text{ cm})(1.5 \times 10^2 \text{ cm})} [\tan^{-1} (\frac{300 \text{ cm} + 10 \text{ cm}}{150 \text{ cm}}) - \tan^{-1} (\frac{10 \text{ cm}}{150 \text{ cm}})]$$

$$= (7.33 \times 10^{-1} \frac{R}{hr}) [\tan^{-1} (2.07) - \tan^{-1}(0.07)]$$

where $\tan^{-1} (2.07) = 64.22$
and $\tan^{-1} (0.07) = 3.81$ [$\tan^{-1}$ = arc tangent]

$$ER = (7.33 \times 10^{-1} \frac{R}{hr}) [64.22 - 3.81]$$

$$= (7.33 \times 10^{-1} \frac{R}{hr} (60.41)$$

$$ER = 44.3 \frac{R}{hr}.$$

The exposure rate for sources of other geometries can be calculated. One of the other geometries is the ring-shaped source illustrated in Figure 19-4-10.

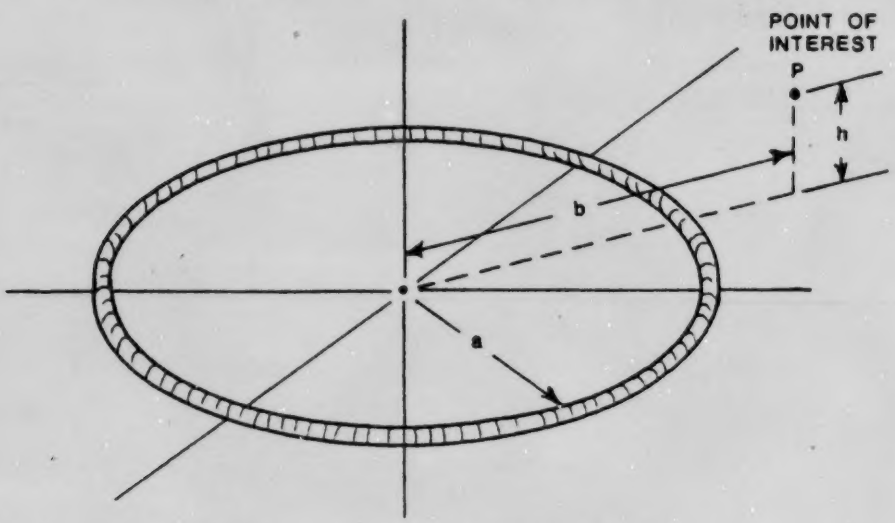


Figure 19-4-10. Ring-shaped source.

For such a ring-shaped source, the exposure rate at a point of interest (P) in Figure 19-4-10 can be calculated from Equation 19-4-3.

$$ER = \frac{\Gamma A}{\sqrt{(a^2 + b^2 + h^2)^2 - 4a^2b^2}} \frac{R}{hr} \qquad \text{Eq. 19-4-3}$$

where

a = ring radium in cm.

b and h are the same as defined in Eq. 19-4-2, except that here b is not measured from the end of the source.

EXAMPLE

Given: A ring source of evenly distributed ^{226}Ra - 100 mCi.

Find: The exposure rate at P where a = 15 cm, b = 2m and h = 50 cm.

Solution: Use Equation 19-4-3.

$$ER = \frac{\Gamma A}{\sqrt{(a^2 + b^2 + h^2)^2 - 4a^2b^2}} \frac{R}{hr}$$

where

$\Gamma = 8.25 \frac{R}{hr} \frac{cm^2}{mCi}$.

A = 100 mCi = 10^2 mCi.

a = 15 cm.

b = 2 m = 200 cm.

h = 50 cm.

$$ER = \frac{8.25 \frac{R}{hr} \frac{cm^2}{mCi} \times 10^2 \text{ mCi}}{[(15 \text{ cm})^2 + (200 \text{ cm})^2 + (50 \text{ cm})^2]^2 - [(4)(15 \text{ cm})^2(200 \text{ cm})^2]}$$

$$= \frac{8.25 \times 10^2 \frac{R}{hr} cm^2}{(4.27 \times 10^4 \text{ cm}^2)^2 - (3.6 \times 10^7 \text{ cm}^4)}$$

$$= \frac{8.25 \times 10^2 \frac{R}{hr} cm^2}{1.83 \times 10^9 \text{ cm}^4 - 3.6 \times 10^7 \text{ cm}^4}$$

$$= \frac{8.25 \times 10^2 \frac{R}{hr} cm^2}{1.79 \times 10^9 \text{ cm}^4}$$

$$= \frac{8.25 \times 10^2 \frac{R}{hr} cm^2}{4.23 \times 10^4 \text{ cm}^2}$$

$$= 1.95 \times 10^{-2} \frac{R}{hr}$$

$$= 1.95 \times 10^{-2} \frac{R}{hr} \times \frac{10^3 \text{ mR}}{R}$$

$$= 1.95 \times 10 \frac{mR}{hr}$$

$$ER = 19.5 \frac{mR}{hr}.$$

A disc source is illustrated in Figure 19-4-11. Although the ring-shaped source is hollow, the disc source is a solid disc of material.

The process for the disc source is somewhat more involved, as indicated by Equation 19-4-4.

$$ER = \frac{\Gamma A}{a} \ln \left[\frac{a^2 - b^2 + h^2 + \sqrt{(a^2 - b^2 + h^2)^2 + 4b^2h^2}}{2h^2} \right] \frac{R}{hr} \qquad \text{Eq. 19-4-4}$$

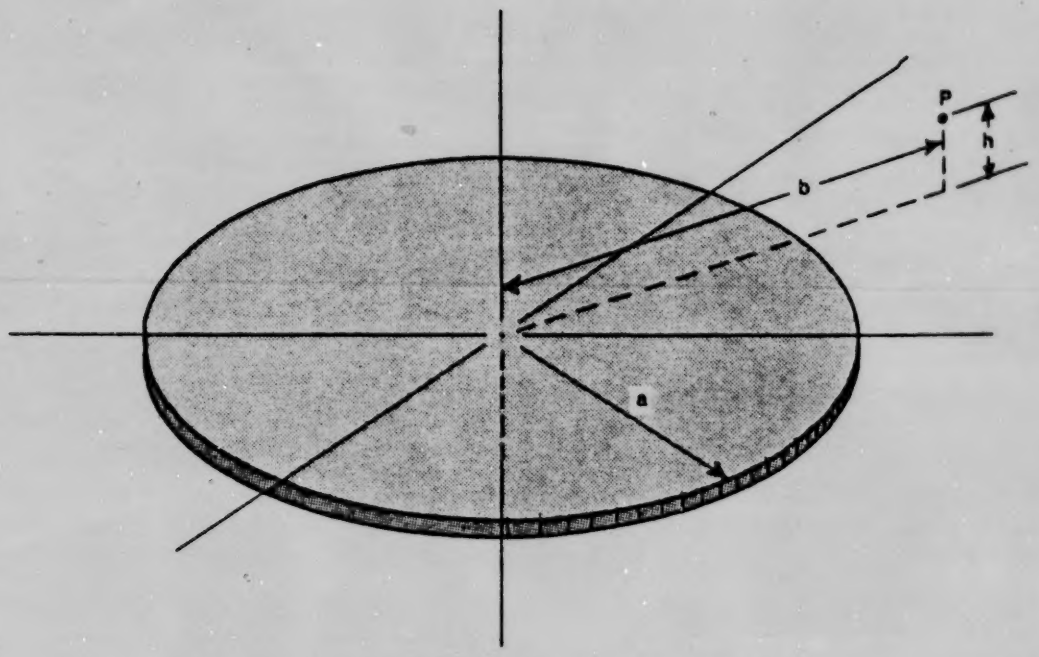


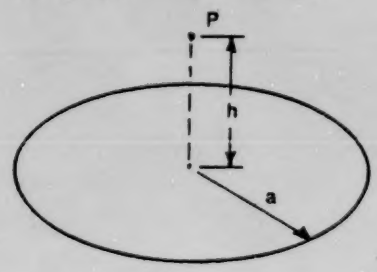
Figure 19-4-11. Disc source.

This equation is utilized in the solving of a problem in the following example.

EXAMPLE

Given: A disc source with ^{60}Co spread evenly over its surface → 5 Ci, radius of disc = a = 1 m.

Find: The exposure rate at P on the central axis (where b = 0) and h = 100 cm.



Solution: Use Equation 19-4-4:

$$ER = \frac{\Gamma A}{a^2} \ln \left[\frac{a^2 - b^2 + h^2 + \sqrt{(a^2 - b^2 + h^2)^2 + 4b^2h^2}}{2h^2}\right] \frac{R}{hr}$$

since

$b = 0.$

$$ER = \left(\frac{\Gamma A}{a^2}\right) \ln \left[\frac{a^2 + h^2 + \sqrt{(a^2 + h^2)^2}}{2h^2}\right],$$

where

$\Gamma = 13.2 \frac{R}{hr} \frac{cm^2}{mCi}.$

$A = 5\ Ci = 5 \times 10^3\ mCi.$

$a = 1\ m = 100\ cm.$

$h = 100\ cm.$

$$ER = \frac{(13.2 \frac{R}{hr} \frac{cm^2}{mCi})(5 \times 10^3\ mCi)}{(100\ cm)^2} \ln$$

$$\left[\frac{(100\ cm)^2 + (100\ cm)^2 + \sqrt{[(100\ cm)^2 + (100\ cm)^2]^2}}{2(100\ cm)^2}\right]$$

$$= 6.6 \frac{R}{hr} \times \ln \left[\frac{4 \times 10^4\ cm^2}{2 \times 10^4\ cm^2}\right]$$

$$= 6.6 \frac{R}{hr} \times \ln [2] = 6.6 \frac{R}{hr} \times 0.69$$

$$ER = 4.57 \frac{R}{hr}.$$

GEOMETRY OF BEAMS

The "geometry" of a beam of gamma radiation is simply its shape. A beam has "good geometry" when the primary beam is well defined and well collimated (made straight and narrowed).

The shielding setup in Figure 19-4-12 illustrates how shielding can collimate a beam from a source. This arrangement allows the shield to absorb scattering, a major source of error in measurements. Since scattering is reduced and almost eliminated, readings will be relatively accurate.

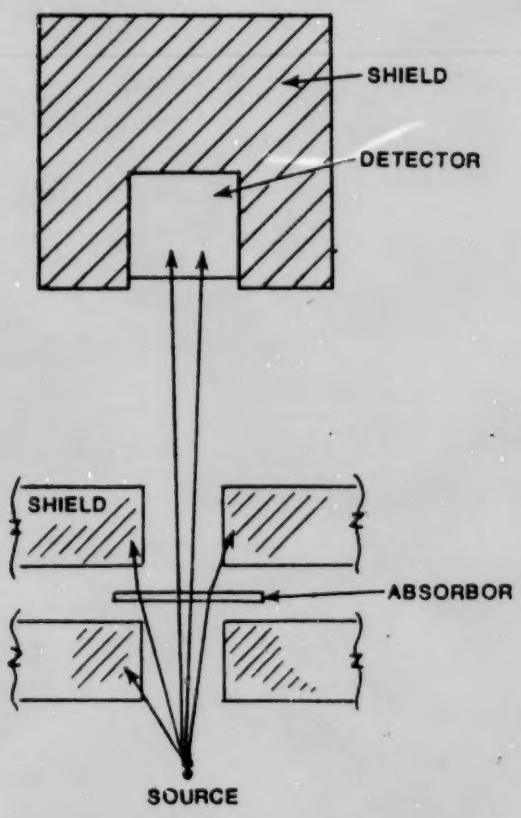


Figure 19-4-12.
Good geometry.

The geometry arrangement in Figure 19-4-13 will cause the detector to overestimate the activity of a sample and, therefore, is an example of "poor geometry." In this setup, radiation scattered from the absorber and from the shield strikes the detector. Technicians often encounter these

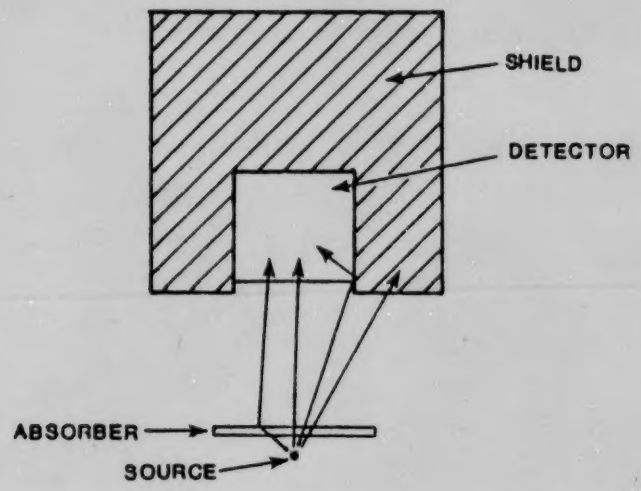


Figure 19-4-13.
Poor geometry.

"poor geometry" situations. Usually, the primary gamma beam is not collimated; radiation is scattered; and a correction factor must be applied.

BUILD-UP FACTOR

Corrections for the effect of scattering must take into account the build-up factor (B). This factor is the ratio of activity measured by the detector (intensity of the primary beam and scatter radiation) to the calculated actual intensity of the primary beam.

When this factor is included, the intensity (I) of gamma rays passing through an absorber of thickness x may be calculated from Equation 19-4-5, a modification of the original attenuation equation.

$$I = I_o e^{-\mu x} \qquad \text{Eq. 19-4-5}$$

where

I_o = initial intensity of gamma rays in units of $\frac{\text{energy}}{(\text{area})(\text{time})}$.

e = base of natural logarithms.

μ = linear absorption coefficient of absorber for given gamma rays in reciprocal centimeters.

x = thickness of absorber in centimeters.

This equation is modified to form Equation 19-4-6.

$$I = BI_o e^{-\mu x} \qquad \text{Eq. 19-4-6}$$

where

B = build-up factor.

Build-up factors can be found on pages 145 through 147 of the Radiological Health Handbook. These build-up factors are for point sources, from which radiation is emitted in all directions (point isotropic sources); and for plane sources, from which radiation is emitted in only one direction. Table 19-4-2 contains dose build-up factors for several materials when used as point isotropic sources.

The inclusion of the build-up factor in the attenuation equation makes shielding calculations more accurate, although the actual calculation of shielding requirements is somewhat more complicated. Since the thickness x must be known before B can be determined, the equation $I = BI_o e^{-\mu x}$ contains two unknowns – B and x. The thickness x must be found by substituting for x in the equation, finding the build-up factor, solving for intensity, and then checking to deter-

TABLE 19-4-2

DOSE BUILD-UP FACTORS

Point Isotropic Source

Material	MeV	μx*						
		1	2	4	7	10	15	20
Tin	0.5	1.56	2.08	3.09	4.57	6.04	8.64	--
	1.0	1.64	2.30	3.74	6.17	8.85	13.7	18.8
	2.0	1.57	2.17	3.53	5.87	8.53	13.6	19.3
	3.0	1.46	1.96	3.13	5.28	7.91	13.3	20.1
	4.0	1.38	1.81	2.82	4.82	7.41	13.2	21.2
	6.0	1.26	1.57	2.37	4.17	6.94	14.8	29.1
	8.0	1.19	1.42	2.05	3.57	6.19	15.1	34.0
	10.0	1.14	1.31	1.79	2.99	5.21	12.5	33.4
Tungsten	0.5	1.28	1.50	1.84	2.24	2.61	3.12	--
	1.0	1.44	1.83	2.57	3.62	4.64	6.25	(7.35)
	2.0	1.42	1.85	2.72	4.09	5.27	8.07	(10.6)
	3.0	1.36	1.74	2.59	4.00	5.92	9.66	14.1
	4.0	1.29	1.62	2.41	4.03	6.27	12.0	20.9
	6.0	1.20	1.43	2.07	3.60	6.29	15.7	36.3
	8.0	1.14	1.32	1.81	3.05	5.40	15.2	41.9
	10.0	1.11	1.25	1.64	2.62	4.65	14.0	39.3
Lead	0.5	1.24	1.42	1.69	2.00	2.27	2.65	(2.73)
	1.0	1.37	1.69	2.26	3.02	3.74	4.81	5.86
	2.0	1.39	1.76	2.51	3.66	4.84	6.87	9.00
	3.0	1.34	1.68	2.43	2.75	5.30	8.44	12.3
	4.0	1.27	1.56	2.25	3.61	5.44	9.80	16.3
	5.1097	1.21	1.46	2.08	3.44	5.55	11.7	23.6
	6.0	1.18	1.40	1.97	3.34	5.69	13.8	32.7
	8.0	1.14	1.30	1.74	2.89	5.07	14.1	44.6
	10.0	1.11	1.23	1.58	2.52	4.34	12.5	39.2
Uranium	0.5	1.17	1.30	1.48	1.67	1.85	2.08	--
	1.0	1.31	1.56	1.98	2.50	2.97	3.67	--
	2.0	1.33	1.64	2.23	3.09	3.95	5.36	(6.48)
	3.0	1.29	1.58	2.21	3.27	4.51	6.97	9.88
	4.0	1.24	1.50	2.09	3.21	4.66	8.01	12.7
	6.0	1.16	1.36	1.85	2.96	4.80	10.8	23.0
	8.0	1.12	1.27	1.66	2.61	4.36	11.2	28.0
	10.0	1.09	1.20	1.51	2.26	3.78	10.5	28.5

*μx = mass absorption coefficient (μ/ρ) x shield thickness (cm) x shield density (g/cm^2).

mine whether or not the intensity is reduced to the desired level. The following example illustrates this process.

EXAMPLE

Given: One-Ci ^{60}Co point source and Radiological Health Handbook.

Find: The thickness of lead necessary to reduce the exposure rate to 2.5 mR/hr at 30.5 cm. Consider buildup.

Solution: General approach: The equation:

$I = I_o B e^{-\mu x}$ had two unknowns, B and x.

To solve, first use $I = I_o e^{-\mu x}$ to approximate x; then substitute x back into $I = I_o B e^{-\mu x}$. Determine B, and see if the approximate value of x is sufficient:

Use $I = I_o e^{-\mu x}$; for Co-60, $\Gamma = 13.2 \frac{R}{hr} \frac{cm^2}{mCi}$.

$C = 1$ Ci

$= 1000$ mCi

$C = 10^3$ mCi,

and $I_o = \Gamma \frac{C}{d^2}$ (Equation 19-4-1)

$= (13.2 \frac{R}{hr} \frac{cm^2}{mCi}) \frac{(10^3 \text{ mCi})}{30.5 \text{ cm})^2} = 14.2 \frac{R}{hr}$

now use

$I = I_o e^{-\mu x}$ to find X,

where

$I = 2.5 \frac{mR}{hr}$, $I_o = 14.2 \frac{R}{hr}$

$$I = 14.2 \times 10^3 \frac{mR}{hr}.$$

Find μ from mass attenuation coefficient and density of lead from

$$\mu(cm^{-1}) = \mu_m \left(\frac{cm^2}{g}\right) \rho \left(\frac{g}{cm^3}\right),$$

where μ_m = mass attenuation coefficient.

From tables in the Radiological Health Handbook (page 138), find mass attenuation coefficients for lead, and for any energy of ~1.25 MeV. Interpolate fo find $\mu_m \tilde{=} 0.06 \frac{cm^2}{g}$.

Therefore, $\mu = (0.06 \frac{cm^2}{g})(11.35 \frac{g}{cm^3})$

$= 0.68/cm.$

Use $I = I_o e^{-\mu x}$ to obtain

$$2.5 \frac{mR}{hr} = (1.42 \times 10^3 \frac{mR}{hr}\ e^{-(0.68/cm)\ (x)}.$$

Solve for x:

$$e^{(0.68/cm)\ (x)} = 5.68 \times 10^3.$$

Take the natural log of both sides (natural log = ln).

$$\ln\left(e^{(0.68/cm)\ (x)}\right) = (0.68/cm)\ (x)$$
$$= \ln\ (5.68 \times 10^3)$$
$$= 8.64$$

$$\text{and } x = \frac{8.64}{0.68/cm} = 12.7 \text{ cm}.$$

Now use this value of x = 12.7 cm and μ = 0.68/cm and the dose build-up factors tables for B in the Radiological Health Handbook (Point Source), page 146, to find B: $\mu x = 8.64$. Interpolate on both μx and energy (1.25 MeV) to find $B \tilde{=} 3.5$.

Now use B in $I = I_o B e^{-x}$ to determine if estimate of x is adequate:

$$I = (14.2 \times 10^3 \frac{mR}{hr})(3.5)\ e^{-(0.68/cm)(12.7\ cm)}$$

$$I = (4.97) \times 10^4 \frac{mR}{hr})\ e^{-8.64}$$

$$= (4.97 \times 10^4 \frac{mR}{hr})(1.78 \times 10^{-4})$$

$$I = 8.83 \frac{mR}{hr},$$

This value of $I > 2.5 \frac{mR}{hr}$; so thickness is not great enough so add one HVL to thickness:

$$HVL = \frac{\ln 2}{\mu} = \frac{0.693}{0.68/cm} = 1.02\ cm$$

$$12.7\ cm + 1.02\ cm = 13.72\ cm,$$

Now recalculate B: $B \approx 3.6$.

Use

$$I = I_o B e^{-\mu}$$

$$= (14.2 \times 10^3 \frac{mR}{hr})(3.6)\ e^{-(0.68/cm)(13.7\ cm)}$$

$$= (5.1 \times 10^4 \frac{mR}{hr})\ e^{-9.32}$$

$$= (5.1 \times 10^4 \frac{mR}{hr})(9 \times 10^{-5})$$

$I = 4.59 \frac{mR}{hr}$, which is still greater than desired 2.5 mR/hr

Therefore, add one more HVL: 13.7 + 1.02 = 14.72 cm and $B \approx 3.75$.

$$I = (14.2 \times 10^3 \frac{mR}{hr})(3.75) e^{-(0.68/xm)(14.7\ cm)}$$

$$= (5.33 \times 10^4 \frac{mR}{hr})\ e^{-9.99}$$

$= (5.33 \times 10^{4} \frac{mR}{hr})(4.56 \times 10^{-5})$

$I = 2.43 \frac{mR}{hr}$.

Thus, 14.72 cm of lead is adequate and will reduce the exposure rate to a value less than 2.5 mR/hr.

GAMMA AND X-RAY SHIELDING

SHIELDING FACTORS

The amount of shielding required for gamma and X-radiation is a function of the maximum permissible dose limit, occupancy factor, use factor, and workload. All of these factors can be found in Appendix C of the Report Structural Shielding Design and Evaluation for Medical Use of X-rays and Gamma Rays of Energies Up To 10 MeV (refer to Reference Materials section of this module).

The maximum permissible dose limits (MPD) for controlled areas is 5,000 mrem/year. Controlled areas are areas in which occupationally-exposed individuals work. For noncontrolled areas – these occupied by persons who are – not occupational radiation workers – the MPD is 500 mrem per year. Once the MPD is known, the hourly exposure rate limit can be calculated from Equation 19-4-7.

$$\text{Hourly exposure rate} = \frac{\text{yearly exposure rate}}{12 \times \text{number of hours worked per month}} \qquad \text{Eq. 19-4-7}$$

EXAMPLE

Given: A MPD of 5,000-mrem/yr and a 168-hour work month.

Find: Hourly exposure-rate limit.

Solution: Let 1 mrem = 1 mR.

From Equation 19-4-7:

$$\text{Hourly exposure rate} = \frac{\text{yearly exposure rate}}{12 \times \text{number of hours worked per month}}$$

$$= \frac{5{,}000}{12 \times 168}$$

$\therefore$ Hourly exposure rate = 2.5 mR/hr.

This exposure-rate limit often is used for I in gamma-shielding calculations for controlled areas (meaning areas in which occupational exposure of personnel to radiation is under the supervision of the Radiation Protection Supervisor).

The occupancy factor (T) for controlled areas is 1. T = 1, 1/4, or 1/16 for noncontrolled areas, depending upon the degree of occupancy.

Table 19-4-3 lists use factors for primary protective barriers. These values are used only if specific values for a given installation are not available. A primary protective barrier shields against the effects of direct beams of gamma radiation.

TABLE 19-4-3

USE FACTORS FOR PRIMARY PROTECTIVE BARRIERS[a]

	Radiographic Installations	Therapy Installations
Floor	1	1
Walls	1/4	1/4
Ceiling	b	c

[a]The use factor for secondary protective barriers is usually 1.

[b]The shielding requirements for the ceiling of a radiographic installation are determined by the secondary barrier requirements rather than by the use factor, which usually is extremely low.

[c]The use factor for the ceiling of a therapy installation depends upon the type of equipment and techniques used, but usually is not more than 1/4.

"Workload" is the amount of time a beam is used. Some typical medical workloads are given in Table 19-4-4. For gamma-emitting sources, the workload usually is expressed in units of MA-min/weeks.

CALCULATION OF SHIELDING REQUIREMENTS

Shielding requirements for gamma radiation can be calculated as follows:

- By use of the attenuation equation (Eq. 19-4-6).
- By use of attenuation curves for selected nuclides.
- By use of a Table of Half-Value Layers (HVL).

TABLE 19-4-4

TYPICAL WEEKLY WORKLOADS FOR BUSY INSTALLATIONS

Diagnostic	Daily Patient Load	Weekly workload (W) mA min		
		100 kV[a] or less	125 kV[a]	150 kV[a]
Admission chest: (Miniature, with photo-timing grid)	100	100	--	--
Chest: (14 x 17: 3 films per patient, no grid)	60	150	--	--
Cystoscopy	8	600	--	--
Fluoroscopy including spot filming	24	1,500	600	300
Fluoroscopy without spot filming	24	1,000	400	200
Fluoroscopy with image intensification including spot filming	24	750	300	150
General radiography	24	1,000	400	200
Special procedures	8	700	280	140
Therapy		**Weekly workload (W)**		
Superficial (up to 150 kV[a])	32	3,000 mA min		
Orthovoltage (200-300 kV[a])	32	20,000 mA min		
Megavoltage (0.5 MV - 10 MV)	50	100,000 R at a meter[b]		
Cesium				
50 cm SSD	16	8,000 R at a meter[b]		
50 cm SSD	32	15,000 R at a meter[b]		
60 cm SSD	32	24,000 R at a meter[b]		
Cobalt				
70 cm SSD	32	30,000 R at a meter[b]		
80 cm SSD	32	40,000 R at a meter[b]		
100 cm SSD	32	60,000 R at a meter[b]		

[a]Peak pulsating X-ray tube potential.

[b]R per week at a meter = R m^2 per week.

The attenuation equation already has been discussed. I always equals the desired level of exposure and is equal to the hourly exposure rate limit in many calculations. Other shielding factors are implicit in the equation.

Attenuation (transmission) curves sometimes provide a shortcut. The transmission of selected radionuclides through lead is depicted on these graphs. Figure 19-4-13 is typical and shows transmission of cobalt-60, cesium-137, and gold-198 through lead. The thickness of lead necessary to reduce radiation to a desired level can be determined as follows:

- Find the factor by which intensity must be reduced.
- Locate that factor on the vertical axis of the graph and the point on the nuclides transmission curve that corresponds to that factor.
- Read down to the horizontal axis to find the necessary thickness of lead.

EXAMPLE

Given: A ^{60}Co source with an exposure rate of 1,000 mR/hr and Figure 19-4-13.

Find: The thickness of lead necessary to reduce this rate to 2 mR/hr.

Solution:
- First find the factor by which the exposure rate must be reached. If 1,000 mR/hr must be reduced to 2 mR/hr, the reduction factor is $\frac{2}{1,000}$, or 2×10^{-3}.
- Find 2×10^{-3} on vertical axis, and index appropriate point on ^{60}Co transmission curve (marked on Figure 19-4-13) by arrows.

- Read down to horizontal axis. About 11.3 cm of lead will be necessary.

Transmission tables are available for other materials besides lead.

HVL tables are available for selected gamma-ray sources; they provide a quick calculation of shielding requirements. Table 19-4-5 is typical.

TABLE 19-4-5
SELECTED GAMMA-RAY SOURCES

Radionuclide	Atomic Number	Half Life	Gamma Energy MeV	Half-Value Layer[a] (HVL)		
				Concrete cm	Steel cm	Lead cm
Cesium-137	55	27 y	0.66	4.8	1.6	0.65
Cobalt-60	27	4.24 y	1.17,1.33	6.2	2.1	1.20
Gold-198	79	2.7 d	0.41	4.1	---	0.33
Iridium-192	77	74 d	0.13 to 1.06	4.3	1.3	0.60
Radium-226	88	1622 y	0.047 to 2.4	6.9	2.2	1.66

[a]Approximate values obtained with large attenuation.

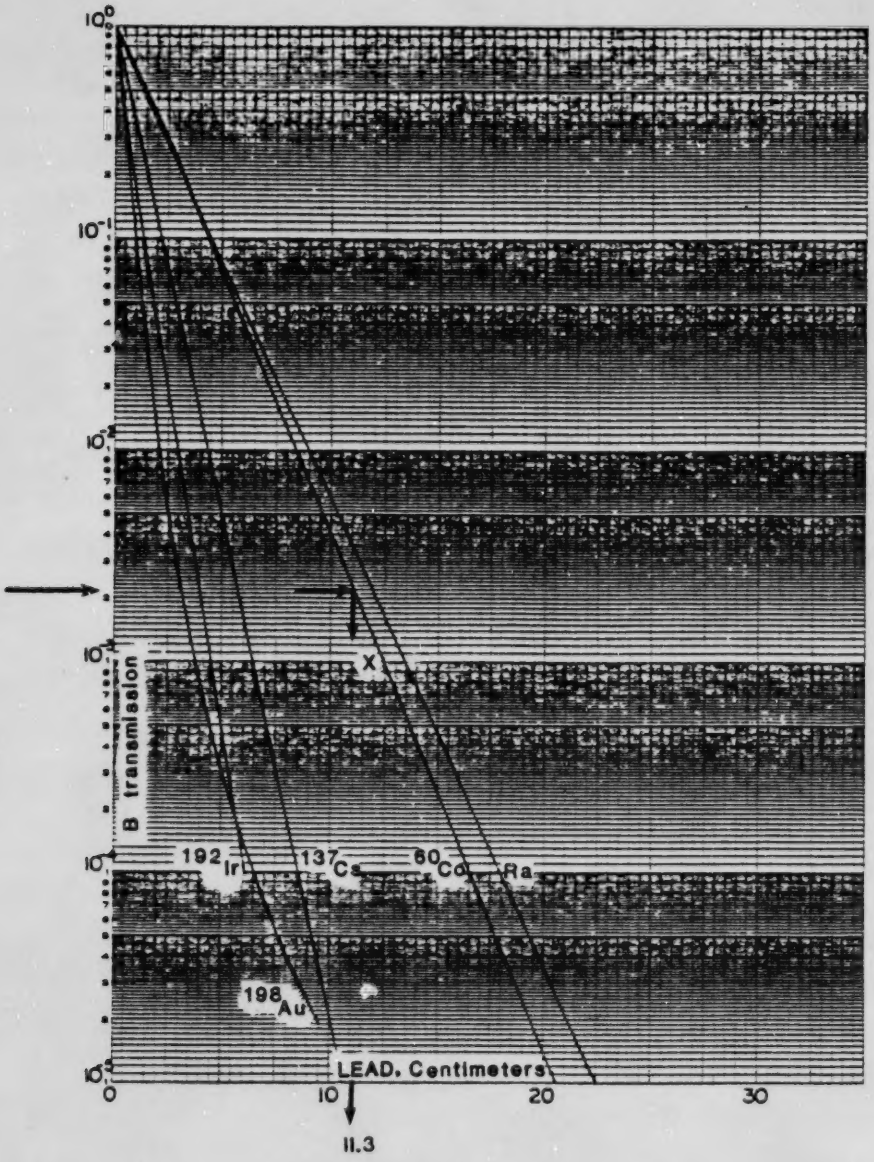


Figure 19-4-13. Transmission through lead of gamma rays from selected radionuclides.

Tables such as Table 19-4-5 list the value for the half-value layer (HVL) of the absorbing material used with various radionuclides. Each additional HVL will reduce the exposure rate by one-half. HVLs are added until the exposure rate is reduced to the desired level, as indicated by the example that follows.

EXAMPLE

Given: A ^{60}Co source with an exposure rate of 1,000 mR/hr and Table 19-4-5.

Find: Thickness of lead necessary to reduce this rate to 2 mR/hr.

Solution:

1. Find HVL for lead (1.20 cm) from Table 19-4-5.
2. As one HVL reduces exposure rate to one-half of its former value, set up table of HVL as follows:

   1 HVL = 500 mR/hr
   2 HVL = 250 0
   3 HVL = 125
   4 HVL = 62.5
   5 HVL = 31.25
   6 HVL = 15.625
   7 HVL = 7.812
   8 HVL = 3.906
   9 HVL = 1.953, which is less than the 2 mR/hr required. nine HVLs gives the desired exposure rate.

3. Shielding needed = 9 x 1.2 cm
= 10.8 cm.

MATERIALS FOR GAMMA SHIELDING

Gamma shielding usually is constructed of lead, iron, steel, sand, or concrete. The actual material used depends upon many factors, including economics.

Because of its low cost and convenience, concrete often is employed as shielding. Users of this material, however, must ensure that the concrete they use possesses the density required. The density of concrete varies according to the type of concrete and the manner in which it is poured.

Whenever possible, primary beam shielding is arranged in such a manner that radiation is directed at low-occupancy areas. This precaution reduces radiation risks, shielding needs, and the cost of shielding. Table 19-4-6 lists the density of some typical building materials.

TABLE 19-4-6

DENSITIES OF SOME TYPICAL BUILDING MATERIALS

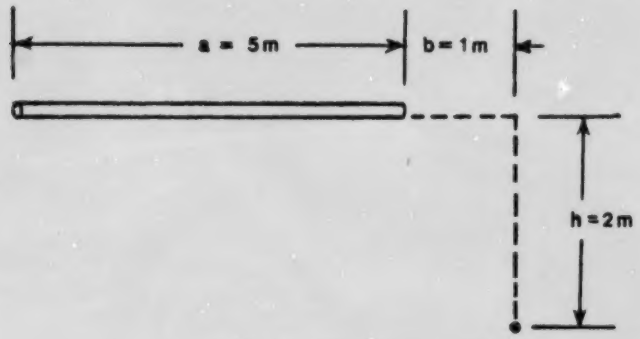
Material	Range of Density[a]	Average Density[b]	
		g cm^{-3}	lb ft^{-3}
Barium sulfate (natural barite)	--	4.5	280
Barytes concrete	3.6 to 4.1	3.6	210
Brick, soft	1.4 to 1.9	1.65	103
hard	1.8 to 2.3	2.05	128
Earth, dry, packed	--	1.5	95
Ferrophosphorus aggregate concrete	6.0	4.8	300
Granite	2.6 to 2.7	2.65	165
Ilmenite aggregate concrete	4.4 to 4.7	3.85	240
Lead	--	11.36	709
Lead glass	--	3.27	205
Lead glass, high density	--	6.22	387
Limestone	2.1 to 2.8	2.46	153
Marble	2.6 to 2.85	2.7	170
Sand, dry, packed	1.6 to 1.9	--	100-120
Sand plaster	--	1.54	96
Concrete	2.25 to 2.4	2.35	147
Steel	--	7.8	489
Tile	1.6 to 2.5	1.9	118

[a]Density values for the concrete aggregates are given for the aggregate only.

[b]Density values are given for the concrete made from the specified aggregate.

Note: Concrete block and cinder blocks vary too much to be listed.

1. Using the specific gamma-ray constant for ^{137}Cs, find the exposure rate for the following configuration:

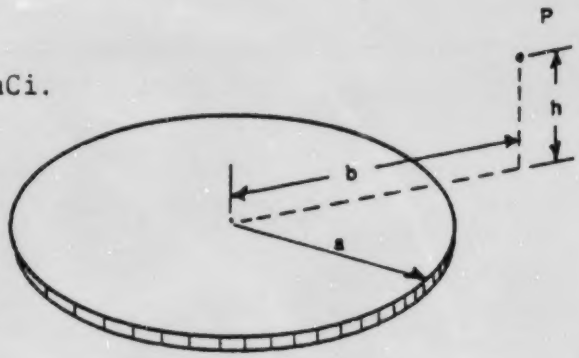


Assume 10 Ci of ^{137}Cs.

2. Find the thickness of lead necessary to reduce the exposure rate from a 10-Ci ^{198}Au point source to 2 mR/hour at 2 meters.

3. Find the exposure rate at a point P, when the source is a disc configuration:

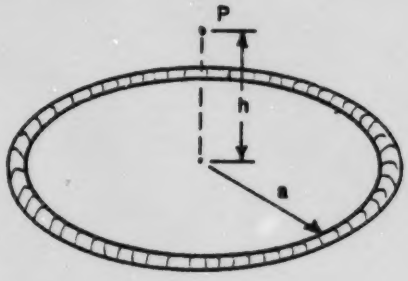
   a = 2 m
   b = 4 m
   h = 5 m
   source: ^{60}Co
   activity: 150 mCi.



4. Find the exposure rate at 10 m from a 5-Ci point source of ^{60}Co.

5. For a ring source of ^{192}Ir, find the exposure rate at point P. P is on the central axis, and thus –

   a = 300 cm
   b = 0
   h = 1 m
   activity = 500 mCi.



MATERIALS

End-window GM survey meter.
^{60}Co source, 10–15 mCi.
Lead bricks.
5 – Lead filters with thickness ranging from 0.5 mm to 6.0 mm.
Meter stick.
Graph paper.

1. Set up bricks for narrow-beam collimation; there should be a 3-mm slit.
2. Measure the instrument response-versus-distance from the source. The distance is measured from the source to the center of the detector. Use Data Table 19-4-1.
3. Plot exposure rate (vertical axis) versus distance (horizontal axis) on graph paper.
4. Remove the lead bricks, and repeat the measurements in Step 2. Use Data Table 19-4-1.
5. Plot exposure rate-versus-distance on the same graph.
6. Explain any difference between the graphs.
7. Now replace the lead bricks for narrow-beam geometry, and record response at a fixed distance (about 30 cm) with different lead filters.
8. From this data, use $I = I_o e^{-\mu x}$, where μm = mass attenuation coefficient (cm^2/g) and where x = density thickness (g/cm^2), to determine μm experimentally. Compare this value to the published value (Radiological Health Handbook) of 0.064 cm^2/g.
9. This procedure is accomplished by a comparison of the exposure rate for any two filters, as indicated by the following example.

EXAMPLE

Given: Filter B gives exposure rate of 3.2 mR/hr, and filter D gives exposure rate of 2.5 mR/hr.

Find: μm.

Solution: For a filter thickness of –

$B = 2300\ mg/cm^2 = 2.3\ g/cm^2$ and for

$D = 6144\ mg/cm^2 = 6.14\ g/cm^2$.

Use $I = I_o e^{-\mu m x}$ (Eq. 19-4-6)

$2.5\ mR/hr = 3.2\ mR/hr\ e^{-\mu m}\ (6.14\ g/cm^2 - 2.3\ g/cm^2)$

To solve for μm –

$$e^{\mu m\ (3.84\ g/cm^2)} = \frac{3.2}{2.5} = 1.28.$$

Take the natural log of each side:

$$\ln\left(e^{\mu m\ (3.84\ g/cm^2)}\right) = \ln\ (1.28)$$

$$\mu m\ (3.84\ g/cm^2) = 0.25$$

$$\mu m = \frac{0.25}{3.84\ g/cm^2}$$

$$\mu m = 0.064\ cm^2/g.$$

DATA TABLE 19-4-1

Distance from Source (cm)	Exposure Rate with Lead Bricks (mR/hr)	Exposure Rate without Lead Bricks
25		
30		
35		
40		
45		
50		
55		
60		
65		
70		
75		
80		
85		
90		

DATA TABLE 19-4-2
RESPONSE AT FIXED DISTANCE

Filter Thickness (g/cm^2)	Exposure Rate (mr/hr)
1	
2	
3	
4	
5	

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1. Describe different types of electromagnetic interactions in matter.

2. From the specific gamma-ray constant, determine the exposure rate from a point source at one meter from one Ci of the following materials:

   ^{60}Co.

   ^{137}Cs.

   ^{192}Ir.

   ^{198}Au.

3. Describe how beam geometry affects the measured activity.

4. Define the following terms:

Rayleigh Scattering.

Thompson Scattering.

Mössbauer Resonance Scattering.

Photodisintegration.

Photoelectric Effect.

Compton Effect.

Pair Production.

Build-up Factor.

5. Calculate the thickness of lead necessary to reduce the exposure rate from a 500-mCi ^{137}Cs source to 2 mR/hr at one meter. Include buildup.

6. Find the ER from a 2-mCi point source of ^{226}Ra at a distance of 5 cm.

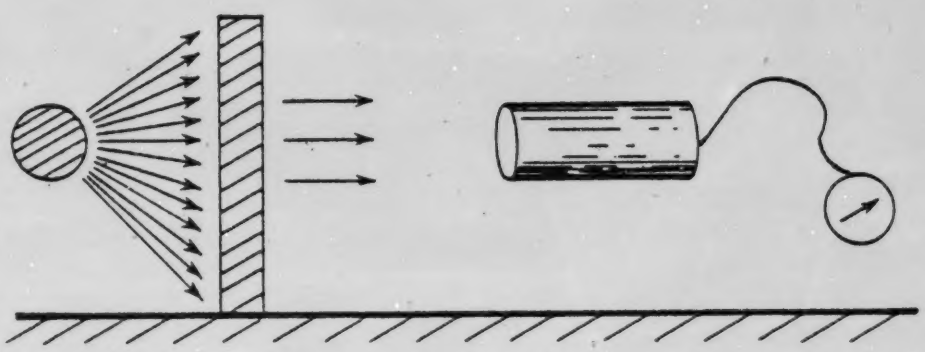
7. Find the ER for the following disc configuration:
a = 3 m, b = 10 m, h = 1 m, Source = ^{192}IR, Activity = 100 mCi.

8. Find the ER for the following ring configuration:
a = 3 m, b = 10 m, h = 1 m, Source = ^{192}IR, Activity = 100 mCi.

NUCLEAR TECHNOLOGY

MODULE 19 - 5

BACKGROUND RADIATION REDUCTION TECHNIQUES



RADIATION SHIELDING

COURSE 19

ORD CENTER FOR OCCUPATIONAL RESEARCH AND DEVELOPMENT

COURSE MODULES

COURSE 19 RADIATION SHIELDING

MODULE 19-1 General Effect of Radiation Attenuation
MODULE 19-2 Beta and Alpha Particle Attenuation
MODULE 19-3 Neutron Attenuation
MODULE 19-4 Gamma Attenuation
MODULE 19-5 Background Radiation Reduction Techniques
MODULE 19-6 Backscattering and Wall Scattering of Radiation

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INTRODUCTION

Background radiation is the radiation emitted from radioactive material other than the one directly under measurement. Background radiation due to cosmic rays and natural radioactivity always is present. The presence of radioactive substances in other parts of the building, in the building material itself, etc., contribute to background radiation.

All counts that result from background radiation must be subtracted from the gross count to obtain the true activity of a sample. The counting process is simplified and made more accurate when background is reduced.

Radiation shielding not only is used for the protection of persons from the unwanted effects of ionizing radiation, but also for the reduction of background radiation in laboratory counting systems. Such reduction increases the efficiency of detector systems by minimizing the effect of background and, consequently, by increasing the accuracy of the measurements.

MODULE PREREQUISITES

The student should have completed a course in basic algebra; Course 1, "Radiation Physics"; Course 2, "Radiation Protection I"; and Course 11, "Radiation Detection and Measurement."

OBJECTIVES

Upon completion of this module, the student should be able to:

1. List and describe two broad categories of background radiation.
2. State the average annual whole-body dose for the following:
   - All natural background radiation.
   - Cosmic radiation.
   - Terrestrial radiation.
   - Radiation from nuclear facilities.
   - Radiation from nuclear testing.
   - Natural radioactivity within the human body.
3. State energy ranges for the following:
   - Galactic cosmic rays.
   - Cosmic rays originating in the solar system.
4. Describe the effect of the geomagnetic field on primary cosmic rays.
5. Describe the relationship of geomagnetic latitude to the threshold energy of cosmic rays.
6. Describe the relationship of altitude to the energy flux of cosmic rays.
7. List the nuclides that have the largest effect on natural radioactivity.
8. Describe the geological formations likely to contain larger-than-average radioactive element concentrations.
9. List radioactive materials that contribute to natural radioactivity within the human body.
10. Calculate shielding requirements necessary for the reduction of background radiation to specified levels.

SOURCES OF BACKGROUND RADIATION

There are two broad categories of background radiation: natural and manmade. Natural background includes cosmic and terrestrial radiation – primarily cosmic rays and radiation from naturally-occurring radionuclides. Natural background accounts for an average dose of 102 mrem per year. Manmade background includes a variety of sources, including radiation from medical facilities; nuclear installations, including uranium mills, power-generating plants, fuel fabrication facilities, and fuel reprocessing plants; releases from nuclear weapons testing; and miscellaneous sources, such as wastes from nuclear-powered ships, nuclear rocket-engine tests, and radiation from television and other electronic products. Even coal-powered plants emit significant amounts of radiation because of the presence of naturally radioactive materials in the coal. The annual whole-body dose rate from both natural and manmade radiation averages about 182 mrem/year.

COSMIC RADIATION

Cosmic radiation contributes an average dose of 44 mrem per year. It includes both the primary radiation from high-energy particles, mainly photons, that are incident upon the earth from outer space, as well as the secondary radiation produced by particles as a result of interaction with atomic nuclei. Secondary particles are generated principally by

the impact of primary radiation against air nuclei at the top of the atmosphere.

Most primary cosmic radiation is made up of highly energetic protons and alpha particles. Most of these particles exhibit energy greater than 10^9 electron volts (eV), although the energy of some particles reaches 10^{30} to 10^{21} eV. These energies cannot be exceeded by any particles accelerated by manmade devices, although modern colliding-beam accelerators can generate particles that possess energies up to 10^{11} to 10^{12} eV.

Most primary cosmic rays reach the earth from outside the solar system. These are galactic cosmic rays; only a small portion of the primary cosmic rays originates in the solar system. Rays that originate in the solar system have relatively moderate energies (less than 10^9 eV) and are associated with the activity of the sun. Rays with greater energies are released during brief, intense upsurges of solar activity. These increases in intensity are associated with solar flares. During some of these flares, the fluxes of solar cosmic rays are hundreds of times more powerful than the fluxes of galactic cosmic rays, and pose a serious threat to manned space flights.

The total energy flux incident on the earth from cosmic rays is about 0.01 erg/cm^2 per second, which is extremely small and is comparable to the energy of the visible radiation of stars. The flux comes from all directions in space and is said to be isotopic (independent of direction). Upon entering the earth's magnetic field, the charged particles of cosmic radiation are deflected from their original direction. Deflection is maximal in equatorial regions and minimal near the poles. This geomagnetic barrier near the equator does not pass protons that have energy less than

~ 15 GeV (giga electron volts). As the geomagnetic latitude increases, the threshold energy of the particles decreases in such a manner that in the polar regions the geomagnetic barrier is negligible. This process results in an increase of average dose by 10 to 20% from 0° to 50° latitude.

Additionally, irregularities in the magnetic field affect the intensity of cosmic rays. Distribution is complex and is illustrated in Figure 19-5-1, in which intensity is given in arbitrary units. The isoclinal lines represent areas of equal intensity of cosmic rays at altitudes in the order of 200 km.

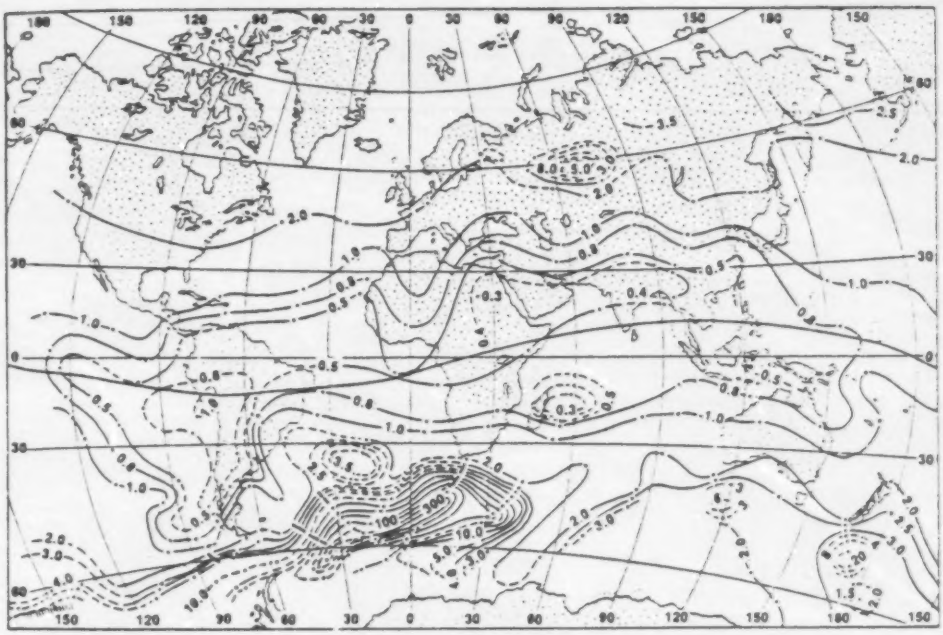


Figure 19-5-1. Chart of isoclinal lines for magnetic fields.

Energy flux also is dependent upon altitude. The average dose of 44 mrem per year increases by a factor of three in going from sea level to an altitude of 10,000 feet. The most elevated inhabited areas of the earth are at an altitude close to 15,000 feet, where cosmic-ray dose is as much as 300 mrem a year. The dose atop Mt. Everest is about 800 mrem per year.

TERRESTRIAL RADIATION

External terrestrial radiation accounts for an average dose of about 58 mrem per year from gamma-emitting nuclides in soil, rock, and construction material.

The radioactive elements in the earth's crust can be divided into two groups – those that have existed since the crust was formed, and those created in recent (geologically speaking) nuclear reactions.

Table 19-5-1 lists some elements that have a half-life comparable with the age of the crust. These nuclides have the largest effect upon general activity.

The second group includes the daughters formed by the decay of the uranium isotopes and thorium. These unstable nuclei have half-lives short on the geological time scale. Although the mass of the daughter elements is extremely small in comparison with that of uranium and thorium, they contribute significantly to the radioactivity of rock. Tables 19-5-2 and 19-5-3 list the ^{238}U and thorium series.

TABLE 19-5-1

ELEMENT HAVING HALF-LIFE COMPARABLE WITH AGE OF CRUST

Nuclear Species	Half-life 10^9 Years	Percentage of Abundance	Heat Generation (joules/ g/year)
^{238}U	4.51	99.27	2.97
^{235}U	0.71	0.72	18.0
^{232}Th	13.9	100.0	0.82
^{40}K	1.3	0.0119	0.92
^{87}Rb	50	27.8	6.7×10^{-3}

The distribution of this radioactive material in the crust affects average dose. In the United States, dose may range from 15 to 35 mrem per year on the Atlantic coast and from 75 to 140 mrem per year on the Colorado plateau. Radioactive element concentrations are higher than average in igneous rocks (rocks formed by intense heat, as by volcanic action), especially near the borders of large intrusions of igneous material. The concentration of uranium and thorium tends to be somewhat higher in granites than in basaltic or ultrabasic material. The abundance of potassium in granites and basalts is approximately 1 percent, and potassium-40 occurs to the extent of 0.012 percent in natural potassium.

The average dose at a height of one yard above limestone is about 20 mrem per year. For granite areas, the corresponding figure is about 150 mrem per year. The dose rate in a building depends upon the radioactive content of the material of which it is made; for example a New Englander living in a house built of granite receives a higher dose

TABLE 19-5-2

URANIUM ^{238}U SERIES

	Isotope	Particle Emitted	Particle Energy (Me V)	Half-life
$^{238}_{92}U$	Uranium	α	4.18	4.51×10^9 years
$^{234}_{90}Th$	Thorium	β	0.205 0.111	24.1 days
$^{234}_{91}Pa$	Proactinium	β	2.32, 1.50, 0.60	1.75 min
$^{234}_{92}U$	(Ruanium)	α	4.763	2.48×10^5 years
$^{230}_{90}Th$	(Thorium or Ionium)	α	4.68, 4.61	8.0×10^4 years
$^{226}_{88}Ra$	(Radium)	α	4.77	1622 years
$^{222}_{86}Rn$	(Radon)	α	5.49	3.83 days
$^{218}_{84}Po$	(Polonium)	α	6.0	3.05 min
$^{214}_{82}Pb$	(Lead)	β	0.65	27 min
$^{214}_{83}Bi$	(Bismuth)	0.04% α 99.96% β	5.46 1.65, 3.17	19.7 min
$^{214}_{84}Po$	(Polonium)	α	7.68	163 μsec
$^{210}_{81}Tl$	(Thallium)	β	1.8	1.3 min
$^{210}_{82}Pb$	(Lead)	β	0.018	22.5 years
$^{210}_{83}Bi$	(Bismuth)	β	1.17	4.99 days
$^{210}_{84}Po$	(Polonium)	α	5.3	138 days
$^{206}_{82}Pb$	(Lead)			Stable

than a Texan living in a wood-frame house.

Measurements made in the United States indicate that the dose rate of doors varies within 45 to 130 mrem per year. Measurements taken inside houses fall within the

TABLE 19-5-3
THORIUM SERIES

Isotope		Particle Emitted	Particle Energy	Half-life
$^{232}_{90}Th$	(Thorium)	α	3.98	13.9×10^9 years
$^{228}_{88}Ra$	(Radium)	β	0.012	6.7 years
$^{228}_{89}Ac$	(Actinium)	β	1.15	6.13 h
$^{228}_{90}Th$	(Thorium)	α	5.42	1.9 years
$^{224}_{88}Ra$	(Radium)	α	5.68	3.64 days
$^{220}_{86}Rn$	(Radon) or Thoron	α	6.28	54.5 sec
$^{216}_{84}Po$	(Polonium)	α	6.77	0.158 sec
$^{212}_{82}Pb$	(Lead)	β	0.355	10.67 hours
$^{212}_{83}Bi$	(Bismuth)	α β	6.05 2.25	60.5 min
$^{212}_{84}Po$	Polonium	α	8.78	0.29 μsec
$^{208}_{81}Tl$	(Thallium)	β	1.79	3.1 min
$^{208}_{82}Pb$	(Lead)			Stable

range of 29 to 90 mrem. In some areas of the world, such as in the large granite and shale deposits in India and Brazil, dose rates are several times higher. In Brazil, several coastal areas have dose rates of 500 mrem per year, and some 30,000 people are exposed. The Madras States of India, a region 200 kilometers long and several hundred meters wide, produces a dose rate of about 1,300 mrem per year, affecting over 100,000 persons. This dose is the highest level of background radiation to which any humans are exposed.

NATURAL RADIOACTIVITY WITHIN THE HUMAN BODY

The human body is another source of natural background radiation, as small traces of radioactive materials normally are present. Radioactive thorium, radium, and lead can be detected, but the dose rates are extremely low (probably less than 1 mrem per year); however, potassium-40 does make a larger contribution to natural background. Although concentrations vary with age and other circumstances, the average annual dose is estimated to be 18 – 20 mrem.

MANMADE RADIOACTIVITY

Manmade background radiation is generated by widespread medical uses of radiation and by nuclear and electronic equipment.

- Medical and dental exposures by diagnostic and therapeutic radiation and radiopharmaceuticals contribute to individual dose and are responsible for about 75 mrem per year.
- Nuclear fuel, fabrication, power, and reprocessing facilities account for about 0.003 mrem per year.
- Nuclear testing formerly released 4 mrem from ^{90}Sr; ^{137}Cs; and gaseous nuclides, such as ^{3}H, ^{14}C, and ^{85}KR. This kind of testing is not done now.
- Occupational radiation exposure results in an average annual dose of about 0.8 mrem per year, but not to the general public.
- Miscellaneous sources of background exposure include wastes from nuclear-powered ships, nuclear rocket-engine testing, air transportation, and television

and other electronic products which together account for about two mrem per year.

Table 19-5-4 lists annual whole-body dose rates from both natural and manmade sources.

TABLE 19-5-4
ANNUAL WHOLE-BODY DOSE RATES

Source	Average Dose Rate (mrem/year)
Cosmic radiation	44
Terrestrial radiation	58
Medical	73
Nuclear power	0.003
Nuclear testing	4
Occupational exposure	0.8
Miscellaneous	2
	182

EFFECT OF BACKGROUND RADIATION UPON MEASUREMENTS

Background radiation will affect the measurement of radiation from samples by increasing the count rate. Of all sources of background radiation, cosmic radiation and natural radioactivity of the surroundings are primarily responsible for the increased count rate. Background radiation, therefore, will introduce an error in the measurement of activity. In order that the true activity can be

obtained, the background radiation response must be measured and then subtracted from the measurement of sample plus background activity. The gross count and the background count can be counted for the same time interval; but for cases in which the relative magnitude of the source and the background count rates differ greatly, sample-plus-background and background may be counted for different times. When the sample count rate approaches that of the background, then the background should be counted for a longer time in order that the standard error in the count rate can be reduced.

SHIELDING OF BACKGROUND RADIATION

For precision measurements of low-level radiation sources, the counting facility often must be shielded in order to reduce the effects of background radiation. Background radiation usually is the gamma type; therefore, the principles of gamma-ray shielding apply. (Module 19-4, "Gamma Attenuation," discussed these principles in detail.)

The effect of background radiation is reduced if the detector is shielded. For a GM tube, 5.02 centimeters of lead reduces the background activity by a factor of about four.

The most common shielding materials are lead, steel, mercury, and water. Since contamination of the material may be a problem, pure water as a shield has the advantage that, although it is a low-Z material, large quantities of it can be prepared.

Scattering and attenuation characteristics of the shielding material must be considered. When used to atten-

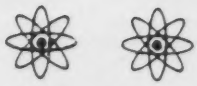
uate incoming gamma rays, steel has a low photoelectric cross section for photons above 100 keV. If no other shielding is employed, some photons in energy range of 100 to 300 keV will interact with the detector.

GRADED SHIELD

A graded shield is constructed of materials of successively lower atomic number used to absorb X-rays from outer layers. A cadmium liner is placed inside a lead shield to absorb the lead X-rays having energies of 73 to 88 keV. The cadmium X-rays of 22 keV are absorbed by a liner of copper or steel. If the outermost shield is steel, a lead liner is followed by a cadmium one and then a copper one.

EXERCISES

Not applicable.



MATERIALS

Gamma-sensitive scintillation micro R meter.
End window GM tube – organic and halogen quenched.
Scintillation detector with scalar.
Shielding material – lead bricks; and iron and aluminum

sheets. (There should be five or six pieces of each material; the pieces should be large enough to fashion an enclosure around the detectors used.)

Masking tape.

PROCEDURES

MEASUREMENT OF BACKGROUND RADIATION

Note: Record data in Data Table 19-5-1.

1. Remove all sources of radiation from the measurement areas.
2. With meter set on most sensitive range, measure the background radiation levels inside of wood buildings and of concrete and/or brick structures. Also measure the background radiation outside of any building. Make five-minute counts.
3. Complete Data Table 19-5-1.

EFFECT OF SHIELDING ON BACKGROUND RADIATION

1. Remove all sources of radiation from the measurement area.
2. With each type of detection instrument, measure the background with no shielding. Make 5-minute counts. Enter results in Data Table 19-5-2.
3. Repeat the measurement with different shielding materials – first aluminum, then iron, and finally lead.

Hold shields in place around detector with masking tape, eliminating as many cracks between joints as possible.

4. Complete Data Table 19-5-2.

DATA TABLES

DATA TABLE 19-5-1

MEASUREMENT OF BACKGROUND RADIATION

Location	Building Material	Exposure Rate
1.		
2.		
3.		
4.		
5.		
6.		
7.		
8.		
9.		
10.		

DATA TABLE 19-5-2

SHIELDING FOR BACKGROUND RADIATION

	Type Detector Used _____			
Detector	Background (cpm) _____			
	No Shielding	Aluminum	Iron	Lead Bricks
#1	_____	_____	____	_____
#2				
#3				

REFERENCE MATERIALS

Cember, H. Introduction to Health Physics. Elmsford, NY: Pergamon Press, 1969.

Chase, Grafton D. and Rabinowitz, Joseph L. Principles of Radioisotope Methodology. 3rd ed. Minneapolis: Burgess Publishing Company, 1967.

Henry, Hugh. Fundamentals of Radiation Protection. New York: Wiley-Interscience, 1969.

Natural Background Radiation in the United States. Report No. 45. Washington, DC: National Council on Radiation Protection and Measurements, 1975.

Nuclear Power and the Environment: Questions and Answers. Hinsdale, IL: The American Nuclear Society, 1974.

The Effects on Populations of Exposure to Low Levels of Ionizing Radiation. Washington, DC: National Academy of Sciences, 1972.

Wang, C. H.; Willis, David L.; and Loveland, Walter D. Radiotracer Methodology in the Biological, Environmental, and Physical Sciences. Englewood Cliffs, NJ: Prentice-Hall, Inc., 1975.

Section I: Matching.

Match the sources of background radiation in column A with the appropriate average annual whole-body dose in column B.

A		B	
1._____	All natural background radiation.	A.	2 mrem.
2._____	Cosmic radiation.	B.	0.8 mrem.
3._____	Radiation from nuclear facilities.	C.	44 mrem.
4._____	Radiation from nuclear testing.	D.	58 mrem.
5._____	Terrestrial radiation.	E.	0.003 mrem.
6._____	Natural radioactivity within the human body.	F.	102 mrem.
		G.	18 – 20 mrem.

Section II: Multiple Choice.

Choose the best answer(s) to the following statements:

7. Energy ranges for galactic cosmic rays are –
   a. $< 10^5$ eV.
   b. $\cong 10^9$ ev.
   c. $> 10^{21}$ eV.
   d. $\cong 10^9$ to 10^{21} eV.
8. Cosmic rays, which originate in the solar system, have energies of –

a. $< 10^3$ eV.
b. $\cong 10^9$ eV.
c. $> 10^{15}$ eV.
d. $\cong 10^{11}$ to 10^{15} eV.

9. When primary cosmic rays encounter the geomagnetic field, they –
   a. Are deflected into a narrow beam and accelerated.
   b. Are deflected maximally in the polar regions and minimally in equatorial regions.
   c. Are not affected, due to their great energies.
   d. Are deflected minimally in the polar regions and maximally in the equatorial regions.

10. From 0° to 50° latitude, average dose from cosmic rays –
   a. Increases by 50%.
   b. Decreases by 10%.
   c. Increases by 10 to 20%.
   d. Remains the same.

11. From sea level to 10,000 feet, the average dose from cosmic rays –
   a. Increases by a factor of three.
   b. Decreases slightly.
   c. Increases by a factor of five.
   d. Remains the same.

Section III: Statement Completion.

Complete the following statements:

12. The nuclides that have the largest effect on natural radioactivity are –

13. Radioactive element concentrations are higher than average in_____.

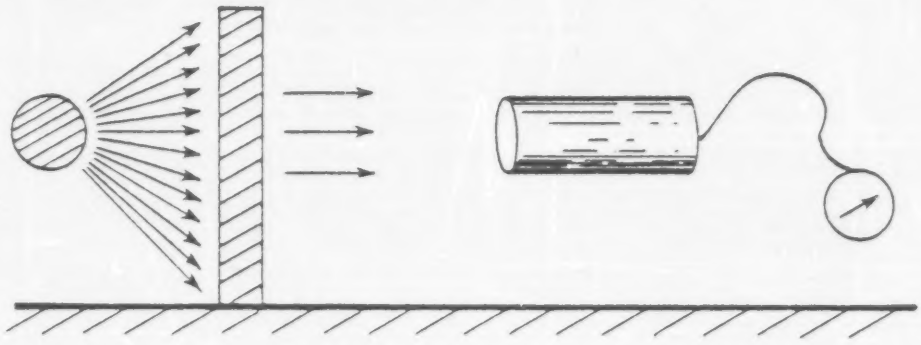
14. The concentrations of uranium and thorium tend to be relatively higher in_____than in basaltic material.

15. The radioactive materials that contribute to natural radioactivity within the human body are –

NUCLEAR TECHNOLOGY

MODULE 19 - 6

BACKSCATTERING AND WALL SCATTERING OF RADIATION



RADIATION SHIELDING

COURSE 19

ORD CENTER FOR OCCUPATIONAL RESEARCH AND DEVELOPMENT

COURSE MODULES

COURSE 19 RADIATION SHIELDING

MODULE 19-1 General Effect of Radiation Attenuation
MODULE 19-2 Beta and Alpha Particle Attenuation
MODULE 19-3 Neutron Attenuation
MODULE 19-4 Gamma Attenuation
MODULE 19-5 Background Radiation Reduction Techniques
MODULE 19-6 Backscattering and Wall Scattering of Radiation

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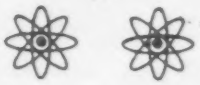
INTRODUCTION

Modules 19-1 through 19-5 have discussed shielding requirements for unique and single types of radiation. Frequently, however, radiation protection shielding must be designed and constructed to shield from mixed radiation fields: combinations of different radiations at varying energies.

This module discusses mixed radiation fields and the shielding commonly used for them. The module also explores special shielding problems, structural aspect of shielding, types of shielding, and shipping containers.



MODULE PREREQUISITES

The student should have completed a course in basic algebra; Course 1, "Radiation Physics"; and Course, "Radiation Protection."

OBJECTIVES

Upon completion of this module, the student should be able to:

1. Specify types of shielding for mixed radiation fields.
2. List three ways in which mixed radiations occur naturally.

3. Write an equation that expresses the relationship between the fraction of the incident beta radiation converted into photons, the atomic number of the absorber, and the maximum energy of the incident beta particles.
4. Specify shielding for situations when bremsstrahlung or X-radiation is stronger than gamma flux and when beta sources have high energies.
5. Discuss the structural aspects of lead and masonry shielding.
6. Calculate the exposure rate due to skyshine at a point, P, given source activity, horizontal distance from point P, and the specific gamma-ray constant of the source.
7. Describe how radioactive material is classified according to form, transport group, and quantity.
8. Describe how a Transport Index is assigned to a package of radioactive material.
9. Describe how the Transport Index is used to establish stowage control.
10. List the types of warning labels, and describe how they are placed on packages of radioactive material.
11. Define the following terms:

    Normal form material.
    Special form material.
    Transport group.
    Type "A" quantities.
    Type "B" quantities.
    Type "A" package.
    Type "B" package.
    Small quantity material.
    Low specific activity material.
    Transport Index.

White label package.
Yellow label package.

Shielding requirements for mixed radiation fields are more complex than those for single types of radiation. Mixed fields occur both naturally (by decay and by interactions in matter) and artificially (when several sources that emit different radiations are stored together).

Mixed fields occur naturally when a single source gives rise to several types and energies of radiation in one of the following three ways.

- The four radioactive chains of thorium, neptunium, uranium, and actinium produce alpha, beta, and gamma radiations that possess great varieties of decay energies. Tables 19-6-1 through 19-6-4 list the major radiation energies and intensities for thorium, neptunium, uranium, actinium, and their daughters. The great variety of radiations and intensities is obvious immediately from these tables; for example, gamma energies in the thorium series (Table 19-6-1) range from 0.34 MeV to 2.614 MeV, and all four series demonstrate a wide variety of both types and intensities of radiation.
- Mixed radiations also occur naturally when neutron radiation produces secondary gamma rays by neutron capture. The production of secondary gamma rays is less important for higher neutron energies. Table 19-6-5 lists the highest-energy gamma rays for materials bombarded by thermal neutrons.
- A third means by which mixed radiation fields occur naturally is X-ray emission through bremsstrahlung caused by an inelastic nuclear collision of a beta particle and a nucleus. In this interaction, the

TABLE 19-6-1

THORIUM SERIES (4n)*

Nuclide	Historical name	Half-life	Major radiation energies (MeV) and intensities† α	β	γ
$^{232}_{90}Th$	Thorium	1.41×10^{10}y	3.95 (24%) 4.01 (76%)	---	---
$^{228}_{88}Ra$	Mesothorium I	6.7y	---	0.055 (100%)	---
$^{228}_{89}Ac$	Mesothorium II	6.13h	---	1.18 (35%) 1.75 (12%) 2.09 (12%)	0.34c‡ (13%) 0.908 (25%) 0.96c (20%)
$^{228}_{90}Th$	Radiothorium	1.910y	5.34 (28%) 5.43 (71%)	---	0.084 (1.6%) 0.214 (0.3%)
$^{224}_{88}Ra$	Thorium X	3.64d	5.45 (6%) 5.68 (94%)	---	0.241 (3.7%)
$^{220}_{86}Rn$	Emanation Thoron (Tn)	55s	6.29 (100%)	---	0.55 (0.07%)
$^{216}_{84}Po$	Thorium A	0.15s	6.78 (100%)	---	---
$^{212}_{82}Pb$	Thorium B	10.64h	---	0.346 (81%) 0.586 (14%)	0.239 (47%) 0.300 (3.2%)
$^{212}_{83}Bi$	Thorium C	60.6m	6.05 (25%) 6.09 (10%)	1.55 (5%) 2.26 (55%)	0.040 (2%) 0.727 (7%) 1.620 (1.8%)
64.0% / 36.0%					
$^{212}_{84}Po$	Thorium C'	304ns	8.78 (100%)	---	---
$^{208}_{81}Tl$	Thorium C"	3.10m	---	1.28 (23%) 1.52 (21%) 1.80 (50%)	0.511 (23%) 0.583 (86%) 0.860 (12%) 2.614 (100%)
$^{208}_{82}Pb$	Thorium D	Stable	---	---	---

*This expression describes the mass number of any member in this series, where n is an integer.

Example: $^{232}_{90}Th$ (4n).....4(58) = 232

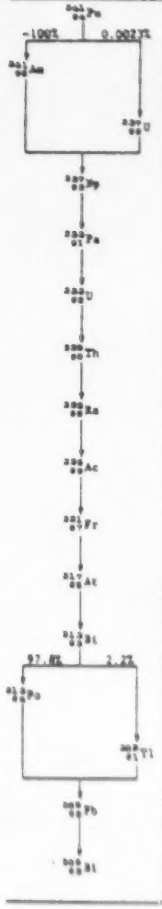
†Intensities refer to percentage of disintegrations of the nuclide itself, not to original parent of series.

‡Complex energy peak which would be incompletely resolved by instruments of moderately low resolving power such as scintillators.

Data taken from: Lederer, C. M., Hollander, J. M., and Perlman, I., Table of Isotopes (6th ed.; New York: John Wiley & Sons, Inc., 1967) and Hogan, O. H., Zigman, P. E., and Mackin, J. L., Beta Spectra (USNRDL-TR-802 [Washington, D.C.: U.S. Atomic Energy Commission, 1964]).

TABLE 19-6-2

NEPTUNIUM SERIES (4n + 1)*



Nuclide	Element name	Half-life	Major radiation energies (MeV) and intensities† α	β	γ
$^{241}_{94}Pu$	Plutonium	13.2y	4.89 (0.0003%) 4.90 (0.001%)	0.021 (~100%)	0.145 (.00016%)
$^{241}_{95}Am$	Americium	458y	5.44 (13%) 5.49 (85%)	---	0.060 (36%) 0.101c‡ (0.04%)
$^{237}_{93}U$	Uranium	6.75d	---	0.248 (96%)	0.060 (36%) 0.208 (21%)
$^{237}_{93}Np$	Neptunium	2.14×10^6y	4.63c (12%) 4.78c (75%)	---	0.030 (14%) 0.086 (14%) 0.145 (1%)
$^{233}_{91}Pa$	Protactinium	27.0d	---	0.145 (37%) 0.257 (56%) 0.568 (5%)	0.31c (44%)
$^{233}_{92}U$	Uranium	1.62×10^5y	4.78 (15%) 4.82 (83%)	---	0.042 (?) 0.097 (?)
$^{229}_{90}Th$	Thorium	7340y	4.84 (58%) 4.90 (11%) 5.05 (7%)	---	0.137c (~3%) 0.20c (~10%)
$^{225}_{88}Ra$	Radium	14.8d	---	0.32 (100%)	0.040 (33%)
$^{225}_{89}Ac$	Actinium	10.0d	5.73c (10%) 5.79 (28%) 5.83 (54%)	---	0.099 (?) 0.150 (?) 0.187 (?)
$^{221}_{87}Fr$	Francium	4.8m	6.12 (15%) 6.34 (82%)	---	0.218 (14%)
$^{217}_{85}At$	Astatine	0.032s	7.07 (~100%)	---	---
$^{213}_{83}Bi$	Bismuth	47m	5.87 (~2.2%)	1.39 (~97.8%)	0.437 (?)
$^{213}_{84}Po$	Polonium	4.2μs	8.38 (~100%)	---	---
$^{209}_{81}Tl$	Thallium	2.2m	---	1.99 (100%)	0.12 (30%) 0.45 (100%) 1.56 (100%)
$^{209}_{82}Pb$	Lead	3.30h	---	0.637 (100%)	---
$^{209}_{83}Bi$	Bismuth	Stable (>2×10^{18}y)	---	---	---

*This expression describes the mass number of any member in this series, where n is an integer.

Example: $^{229}_{90}Th$ (4n + 1).....4(57) + 1 = 229

The (4n + 1) series is included here for completion. It is not found as a naturally-occurring series.

†Intensities refer to percentage of disintegrations of the nuclide itself, not to original parent of series.

‡Complex energy peak which would be incompletely resolved by instruments of moderately low resolving power such as scintillators.

Data taken from: Table of Isotopes and USNRDL-TR-802.

TABLE 19-6-3

URANIUM SERIES (4n + 2)*

Nuclide	Historical name	Half-life	Major radiation energies (MeV) and intensities† α	β	γ
$^{238}_{92}U$	Uranium I	4.51×10^{9}y	4.13 (23%) 4.20 (77%)	---	---
$^{234}_{90}Th$	Uranium X_1	24.1d	---	0.103 (21%) 0.193 (79%)	0.063σ (3.7%) 0.093σ (4%)
$^{234}_{91}Pa^{m}$	Uranium X_2	1.17m	---	2.29 (98%)	0.763 (0.30%) 1.001 (0.60%)
99.87% | 0.13%					
$^{234}_{91}Pa$	Uranium Z	6.75h	---	0.53 (66%) 1.13 (13%)	0.100 (30%) 0.70 (24%) 0.90 (70%)
$^{234}_{92}U$	Uranium II	2.47×10^{5}y	4.72 (28%) 4.77 (72%)	---	0.053 (0.2%)
$^{230}_{90}Th$	Ionium	8.0×10^{4}y	4.62 (24%) 4.68 (76%)	---	0.068 (0.6%) 0.142 (0.07%)
$^{226}_{88}Ra$	Radium	1602y	4.60 (6%) 4.78 (95%)	---	0.186 (4%)
$^{222}_{86}Rn$	Emanation Radon (Rn)	3.823d	5.49 (100%)	---	0.510 (0.07%)
$^{218}_{84}Po$	Radium A	3.05m	6.00 (~100%)	0.33 (~0.019%)	---
99.98% | 0.02%					
$^{214}_{82}Pb$	Radium B	26.8m	---	0.65 (50%) 0.71 (40%) 0.98 (6%)	0.295 (19%) 0.352 (36%)
$^{218}_{85}At$	Astatine	~2s	6.65 (6%) 6.70 (94%)	? (~0.1%)	---
$^{214}_{83}Bi$	Radium C	19.7m	5.45 (0.012%) 5.51 (0.004%)	1.0 (23%) 1.51 (40%) 3.26 (19%)	0.609 (47%) 1.120 (17%) 1.764 (17%)
99.96% | 0.02%					
$^{214}_{84}Po$	Radium C'	164μs	7.69 (100%)	---	0.799 (0.014%)
$^{210}_{81}Tl$	Radium C''	1.3m	---	1.3 (25%) 1.8 (36%) 2.3 (19%)	0.296 (80%) 0.795 (100%) 1.31 (21%)
$^{210}_{82}Pb$	Radium D	21y	3.72 (.000002%)	0.016 (85%) 0.061 (15%)	0.047 (4%)
$^{210}_{83}Bi$	Radium E	5.01d	4.83 (.00007%) 4.89 (.00005%)	1.161 (~100%)	---
~100% | 0.0013%					
$^{210}_{84}Po$	Radium F	138.4d	5.303 (100%)	---	0.803(0.0011%)
$^{206}_{81}Tl$	Radium C''	4.19m	---	1.531 (100%)	---
$^{206}_{82}Pb$	Radium G	Stable	---	---	---

*This expression describes the mass number of any member in this series, where n is an integer.

Example: $^{206}_{82}Pb$ (4n + 2).....4(51) + 2 = 206

†Intensities refer to percentage of disintegrations of the nuclide itself, not to original parent of series.

‡Complex energy peak which would be incompletely resolved by instruments of moderately low resolving power such as scintillators.

Data taken from: Table of Isotopes and USNRDL-TR-802.

TABLE 19-6-4

ACTINIUM SERIES (4n + 3)*

Nuclide	Historical name	Half-life	Major radiation energies (MeV) and intensities†		
			α	β	γ
$^{235}_{92}U$	Actinouranium	7.1×10^8y	4.37 (18%) 4.40 (57%) 4.38c‡ (8%)	---	0.143 (11%) 0.185 (54%) 0.204 (5%)
$^{231}_{90}Th$	Uranium Y	25.5h	---	0.140 (45%) 0.220 (13%) 0.303 (40%)	0.026 (2%) 0.084c (10%)
$^{231}_{91}Pa$	Protoactinium	3.25×10^4y	4.95 (22%) 5.01 (24%) 5.02 (23%)	---	0.027 (6%) 0.29c (6%)
$^{227}_{89}Ac$	Actinium	21.6y	4.86c (0.18%) 4.95c (1.2%)	0.043 (~99%)	0.070 (0.08%)
98.6% 1.4%					
$^{227}_{90}Th$	Radioactinium	18.2d	5.76 (21%) 5.98 (24%) 6.04 (23%)	---	0.050 (8%) 0.237c (13%) 0.31c (8%)
$^{223}_{87}Fr$	Actinium K	22m	5.44 (~0.005%)	1.15 (~100%)	0.050 (40%) 0.080 (13%) 0.234 (4%)
$^{223}_{88}Ra$	Actinium X	11.43d	5.61 (26%) 5.71 (54%) 5.75 (9%)	---	0.149c (10%) 0.270 (10%) 0.33c (6%)
$^{219}_{86}Rn$	Emanation Actinon (An)	4.0s	6.42 (8%) 6.55 (11%) 6.82 (81%)	---	0.272 (9%) 0.401 (5%)
$^{215}_{84}Po$	Actinium A	1.78ms	7.38 (~100%)	0.74 (~.00023%)	---
~100% .00023%					
$^{211}_{82}Pb$	Actinium B	36.1m	---	0.29 (1.4%) 0.56 (9.4%) 1.39 (87.5%)	0.405 (3.4%) 0.427 (1.8%) 0.832 (3.4%)
$^{211}_{85}At$	Astatine	~0.1ms	8.01 (~100%)	---	---
$^{211}_{83}Bi$	Actinium C	2.15m	6.28 (16%) 6.62 (84%)	0.60 (0.28%)	0.351 (14%)
0.28% 99.7%					
$^{211}_{84}Po$	Actinium C'	0.52s	7.45 (99%)	---	0.570 (0.5%) 0.90 (0.3%)
$^{207}_{81}Tl$	Actinium C''	4.79m	---	1.44 (99.8%)	0.897 (0.16%)
$^{207}_{82}Pb$	Actinium D	Stable	---	---	---

*This expression describes the mass number of any member in this series, where n is an integer.

Example: $^{207}_{82}Pb$ (4n + 3).....4(51) + 3 = 207

†Intensities refer to percentage of disintegrations of the nuclide itself, not to original parent of series.

‡Complex energy peak which would be incompletely resolved by instruments of moderately low resolving power such as scintillators.

Data taken from: Table of Isotopes and USNRDL-TR-802.

TABLE 19-6-5

CAPTURE GAMMA RAYS

Target	Thermal (n,γ) Cross Section	Highest Energy Gamma Ray	Average Number of Photons per Capture
	Barns	MeV	
Aluminum	0.235	7.724	2
Antimony	5.7	6.20	
Arsenic	4.3	7.30	
Barium	1.2	9.23	2.7
Beryllium	0.0095	6.814	
Bismuth	0.034	4.17	
Boron-10	3837	0.473[b]	0.94
Cadmium	2450	9.046	4.1
Calcium	0.44	7.83	
Carbon-12	0.0034	4.95	1.3
Chlorine	33.2	8.56	3.1
Chromium	3.1	9.716	2
Cobalt	37.0	7.486	
Copper	3.77	7.914	2.6
Fluorine	0.008	6.600	
Gadolinium	46000	7.78	3.9
Gold	98.8	6.494	3.5
Hydrogen-1	0.332	2.230	1.0
Hydrogen-2	0.00057	6.251	1.0
Indium	196	6.38	3.3
Iron	2.53	10.16	1.7
Lead	0.17	7.38	
Lithium	70.7	7.26	
Magnesium	0.063	11.089	
Manganese	13.2	7.261	2.6
Mercury	380	7.66	3.3
Molybdenum	2.7	9.15	
Nickel	4.8	8.997	
Niobium	1.15	7.19	2.6
Nitrogen-14	0.075	10.833	
Phosphorus	0.190	7.94	
Platinum	8.8	7.920	
Potassium	2.1	9.39	
Praseodymium	11.6	5.83	
Rhodium	156	6.792	
Samarium	5800	7.89	5.6
Scandium	24	8.85	
Selenium	12.3	10.483	
Silicon	0.160	10.599	
Silver	63	7.27	2.9
Sodium	0.534	6.41	2
Strontium	1.21	9.22	
Sulfur	0.52	8.64	
Tantalum	21	6.07	
Thallium	3.4	6.54	
Tin	0.625	9.35	
Titanium	5.8	10.621	
Tungsten	19.2	7.42	
Vanadium	4.96	7.98	2.5
Zinc	1.10	9.51	
Zirconium	0.18	8.66	

[a]These data are taken from Mittelman and Liedtke (1955), BNL (1958), Bartholomew, et al. (1967), Groshev, et al. (1968 and 1969); and BNL (1964).

[b]Gamma ray from excited state of ^{7}Li following (n,α) reaction.

degree of hazard from the resultant radiation is a function of the atomic number of the absorbing material and the energy of the beta particle. As the atomic number of the absorber and the energy of the beta radiation increase, the hazard increases. This relationship is expressed by Equation 19-6-1.

$$f = 3.5 \times 10^{-4} ZE \qquad \text{Equation 19-6-1 (Equation = Eq.)}$$

where

f = the fraction of incident beta radiation converted into photons.

Z = atomic number of the absorber.

E = maximum energy of the beta in MeV.

Thus, for a given shield, the betas emitted by ^{90}Sr will produce more bremsstrahlung than ^{32}P because of the greater energy of the betas emitted by ^{90}Sr.

Apart from these naturally-occurring mixed fields, shielding often must be constructed that will protect against mixed fields created artifically when different types of sources are stored within the same area. A typical example is storage of a gamma source such as ^{60}Co with a neutron source such as ^{252}Cf. Shielding materials must be designed carefully, in order to provide protection against these mixed radiations.

THE SHIELDING OF MIXED RADIATION FIELDS

When mixed radiation fields are made up of gamma, beta, and alpha radiations and when X-radiation is not created by the betas, shielding requirements are simplified because shielding usually must be determined only for the most penetrating radiation. Shielding that is adequate for gamma radiation will be more than adequate for beta and alpha radiation.

When bremsstrahlung or X-radiation is stronger than the gamma flux, the production of these types of radiation can be reduced by a shield of low atomic number, such as aluminum (Z = 13). In cases where the beta source has a high energy, such as the betas produced by ^{90}Sr, a shield of an inner low-Z material shield surrounded by a higher-Z material such as lead is necessary. The inner shield will minimize the bremsstrahlung production and the outer shield will absorb the X-rays that are produced.

SHIELDING MATERIALS

Shielding materials for alpha, beta, gamma, neutron, and X-radiation can be selected as follows:

- Alpha particles can be absorbed by almost any thickness of any material because of their relatively great mass and charge.
- Beta radiation shielding depends upon the energy of the beta emitter. Normally, a small amount of aluminum or lead is adequate. Bremsstrahlung production may require the use of combination shields such as aluminum and lead.

- Lead or concrete often are utilized for X and gamma radiation, although any material of sufficient thickness is adequate.
- Neutron shielding usually is accomplished with water, paraffin, or some other material of low atomic number. These materials do not attenuate gamma radiation well, and additional gamma shielding may be necessary.

The selection of shielding material depends upon several factors:

- Required thickness and density.
- Cost.
- Possibility of multiple use. (The material may serve both structural and shielding purposes.)
- Optical transparency (if required).
- Permanence.
- Quality control requirements.

The most common shielding materials are lead and concrete. Lead is available in several forms. Sheet lead is available in thicknesses from about 0.40 mm to about 1 centimeter. These sheets are not self-supporting and are damaged easily. Also available are lead-lined wallboard, lead-lined lath, lead-lined concrete blocks and lead bricks – all of which are self-supporting.

Concrete often is used because it is a popular and self-supporting material. Concrete has a density of 2.35 g/cm^3. Concrete blocks are available in both standard and high densities. High-density concrete contains materials such as barytes, magnetite, ilmenite, steel, ferrophosphorus, and lead. It is difficult to obtain a uniform density when mixing these materials with concrete.

Other masonry materials such as cinder blocks, bricks, plaster, and marble are similar in composition to concrete, but vary in density. Table 19-6-6 lists the densities of some commercial building materials.

TABLE 19-6-6

DENSITIES OF COMMERCIAL BUILDING MATERIALS

Material	Range of Density[a]	Average Density[b]	
		g cm^{-3}	lb ft^{-3}
Barium sulfate (natural barite)	–	4.5	280
Barytes concrete	3.6 to 4.1	3.6	210
Brick, soft	1.4 to 1.9	1.65	103
hard	1.8 to 2.3	2.05	128
Earth, dry, packed	–	1.5	95
Ferrophosphorus aggregate concrete	6.0	4.8	300
Granite	2.6 to 2.7	2.65	165
Ilmenite aggregate concrete	4.4 to 4.7	3.85	240
Lead	–	11.36	709
Lead glass	–	3.27	205
Lead glass, high density	–	6.22	387
Limestone	2.1 to 2.8	2.46	153
Marble	2.6 to 2.86	2.7	170
Sand, dry, packed	1.6 to 1.9	–	100-120
Sand plaster	–	1.54	96
Concrete	2.25 to 2.4	2.35	147
Steel	–	7.8	489
Tile	1.6 to 2.5	1.9	118

[a] Density values for the concrete aggregates are given for the aggregate only.
[b] Density values are given for the concrete made from the specified aggregate.
Note: Concrete blocks and cinder blocks vary too much to be listed.
Reference: Mark's Mechanical Engineering Handbook, 5th ed. (McGraw-Hill, New York, 1941); excerpts from a table (pp. 522-523), Approximate Specific Gravity and Density.

[a]Density values for the concrete aggregates are given for the aggregate only.

[b]Density values are given for the concrete made from the specified aggregate.

Note: Concrete blocks and cinder blocks vary too much to be listed.

Reference: Marks's Mechanical Engineering Handbook, 5th ed. (McGraw-Hill, New York, 1941); excerpts from a table (pp. 522-523), Approximate Gravity and Density.

SPECIAL CONSIDERATIONS FOR SHIELDING

Several construction details must be considered before the installation of shielding. The shielding must be constructed in such a manner that it is not impaired by joints, openings for ducts, pipes, or service boxes. Doors and windows should be constructed and installed carefully to ensure protection.

Joints of lead sheets should overlap by not less than 1 cm, and walls constructed of blocks or bricks should have mortar of a density equal to that of the blocks. For multiple-course construction, the joints should be staggered.

Steel or iron can be employed when space is limited or when structural strength is important. Water-filled tanks may be used for neutron radiation, but paraffin and oil are unsuitable because they are fire hazards.

SKYSHINE

Gamma sources that have about 30 Ci or more activity and that have gamma radiation energies of 1 MeV or more can cause significant gamma-ray intensity in the shadow of a wall by Compton Scattering in the air above the source. This gamma-ray intensity at a point beyond a protective wall is called "skyshine."

Skyshine from a gamma source can be determined from Equation 19-6-2.

$$ER_{skyshine} = \frac{\Gamma A}{X^2} (10^{-3}) \qquad \text{Eq. 19-6-2}$$

where

$ER_{skyshine}$ = the exposure rate due to skyshine at point P.

X = the horizontal distance from the source to P.

Γ = the specific gamma-ray constant.

A = source activity (Ci).

10^{-3} = the approximate ratio of scatter to incident radiation for a gamma source.

EXAMPLE

Given: A 40-Ci source of ^{60}Co at a distance of 10 meters from point P.

Find: The ER caused by skyshine at P.

Solution: For ^{60}Co, $\Gamma = 1.32 \frac{R}{h} \frac{m^2}{Ci}$

$X = 10$ meters

$A = 40$ Ci.

Use equation 19-6-2:

$$ER_{sky} = \frac{\Gamma A}{X^2} (10^{-3})$$

$$= \frac{(1.32 \text{ R/h } \frac{m^2}{Ci})(40 \text{ Ci})}{(10 \text{ m})^2} (10^{-3})$$

$$= 5.28 \times 10^{-4} \text{ R/h}$$

$$ER_{sky} = 0.528 \text{ mR/h.}$$

The ER caused by skyshine at P is 0.528 mR/h.

SHIPPING CONTAINERS

The Department of Transportation Hazardous Materials Regulations Title 49, Parts 100 to 199 give federal rules for the transportation of radioactive material. These regulations provide the following data:

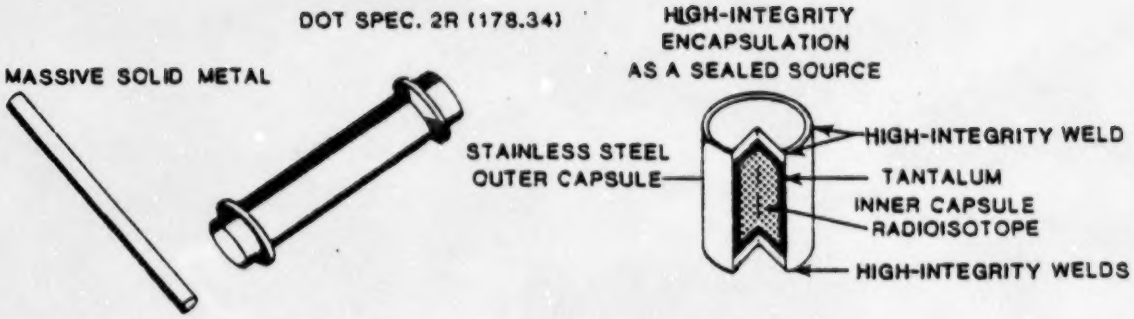
- Definition of radioactive material by quantity, specific activity, and form.
- Classification of radionuclides into one of seven transport groups, based upon their radiotoxicity and potential hazard in transportation.
- Definition of the Transport Index – a number placed on the package of radioactive material to denote the degree of control to be exercised by the shipper.
- Description of the types of packaging.

RADIOACTIVE MATERIAL (RAM)

CLASSIFICATION OF MATERIALS

For packaging and shipping purposes, RAM is classified according to its form, quantity, and type. There are two forms of RAM: Special Form Materials and Normal Form Materials.

Special Form Materials are those that might present some direct radiation hazard if released from a package, but that would present little hazard due to radiotoxicity and little possibility of contamination (Figure 19-6-1).



SPEC. 2R CONTAINMENT DOES NOT AUTOMATICALLY QUALIFY AS SPECIAL FORM. SPECIFIC EVALUATION IS NOT NECESSARY AGAINST 173.398(A).

Figure 19-6-1. Special Form Materials.

Special Form Materials are those that will not spill or scatter easily. They may be encapsuled as a sealed source or in a massive, solid form. Special Form criteria are specified in 49 CFR §173.398 (the symbol § means paragraph). Material must meet or exceed these criteria before it can be categorized as Special Form; for example, it must be able to withstand a drop from a distance of 30 feet, heating in air to a temperature of 1475°F, immersion for 24 hours in water at room temperature, and several other tests, before it can receive the Special Form classification. Once classified as Special Form Material, larger amounts of material can be placed in packages than would otherwise be permitted, allowing more material to be shipped in one package. The classification of material as Special Form is left up to the shipper or package designer. They, of course, must ensure that this material meets federal specifications.

Normal Form Material (Figure 19-6-2) is, simply, any material that does not qualify as Special Form.

WASTE MATERIAL IN PLASTIC BAG

LIQUID IN BOTTLE WITHIN METAL CONTAINER.

NO FIRE RESISTANCE

POWDER IN GLASS OR PLASTIC BOTTLE

LOW MELTING. COMBUSTIBLE AND/OR FRANGIBLE SOLID

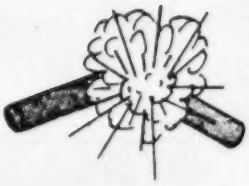


Figure 19-6-2. Normal Form Materials.

TYPE OF MATERIAL

RAM also is classified according to its type. Normal Form Material is placed into one of seven transport groups.

The transport groups are ranged according to their radiotoxicity and their relative potential hazard in transportation. Transport group VII contains the least toxic material; group I, the most toxic. 49 CFR §173.390 lists the transport groups assigned to radionuclides. When radionuclides are mixed, special procedures, outlined in the same paragraph, are used to classify them.

TYPE "A" AND TYPE "B" QUANTITIES

The packaging requirements for RAM are structured around the total quantity of material in a package. This quantity usually is expressed in curies, millicuries, or microcuries. Table 19-6-7 illustrates how quantities are related to packaging requirements.

TABLE 19-6-7
TYPE "A" AND TYPE "B" PACKAGE QUANTITY LIMITS

Transport Group	Type A Quantity (Curies)	Type B Quantity (Curies)
I	0.001	20
II	0.05	20
III	3	200
IV	20	200
V	20	5,000
VI, VII	1,000	50,000
Special Form	20**	5,000

Type "A" quantities can be placed in Type "A" packages; Type "B," in Type "B" packages. A 2-curie amount of a transport group II material, for example, would be classified as a Type "A" quantity and packaged in a Type "A" package.

Figure 19-6-3 illustrates some typical Type "A" packaging schemes. Type "A" packaging must be able to withstand "normal" conditions of transport. These "normal" conditions

are defined in 49 CFR §173.398 (b). These packages also are required to meet the standards for all packages (§173.24) and for specification 7A, general packaging Type "A" (§173.350).

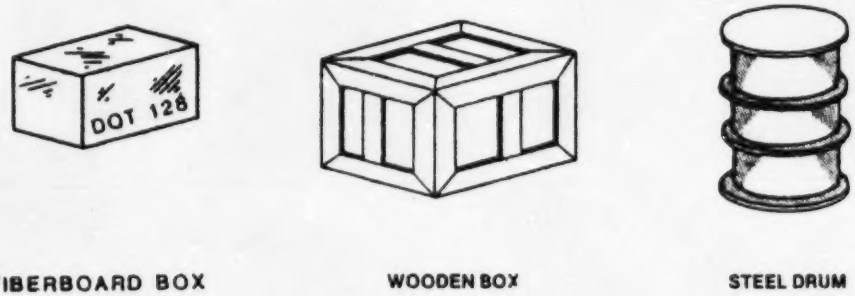


Figure 19-6-3. Typical Type "A" packaging.

Type "A" packaging does not need specific prior approval by DOT, although the shipper of these packages must maintain a certification and safety analysis on file for at least one year.

Type "B" packages must be able to withstand both normal and accidental test conditions without loss of contents (Figure 19-6-4). The performance criteria for the testing of Type "B" packaging are set out in §173.398 (c); they include the following:

- A 30-foot free drop onto an unyielding surface.
- A puncture test, which is a free drop over 40 inches onto a six-inch diameter steel pin.
- Thermal exposure at 1475°F for 30 minutes.
- Water immersion for eight hours (for fissile materials packaging only).

All Type "B" packaging designs require prior approval of the U.S. Nuclear Regulatory Commission.

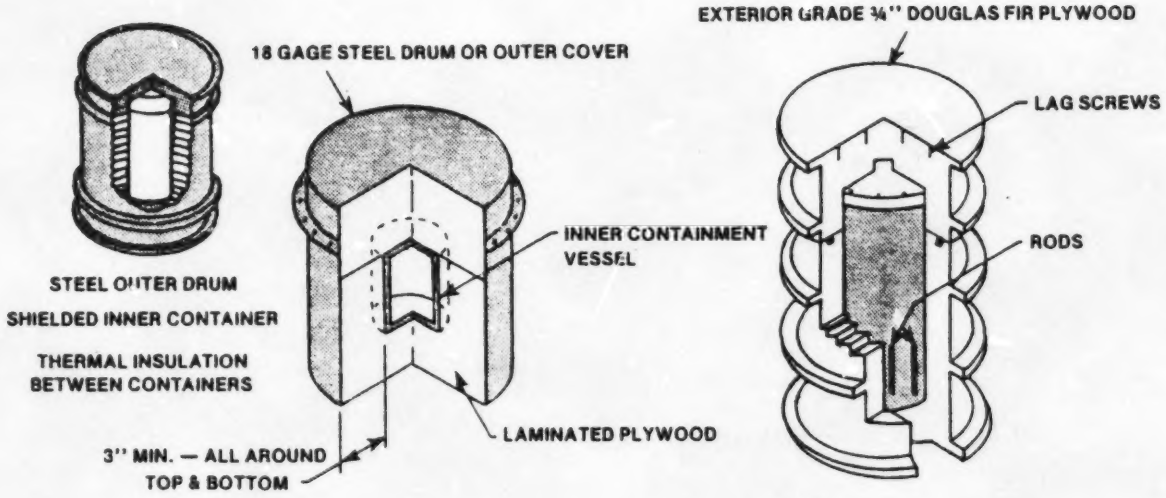


Figure 19-6-4. Typical Type "B" packaging.

LARGE QUANTITIES

Quantities that exceed Type "B" are "Large Quantity" RAM. The most common materials involved as large quantities are the high-curie irradiator sources and irradiated fuel materials. Packaging requirements for these sources involve all the Type "B" packaging requirements plus provisions for decay heat dissipation, heavier shielding, and other requirements (§173.395[c]).

SMALL QUANTITIES

Table 19-6-8 lists the activity limits that a material must meet in order to be qualified as a Small Quantity Mate-

rial (also called "Limited" or "Exempt"). If a RAM is classified as a Small Quantity it may be sent by mail (provided that certain procedures are followed) and is exempt from general packaging requirements in §173.393. Small Quantity Materials must meet the packaging requirements set down in §173.391.

TABLE 19-6-8

ACTIVITY LIMITS FOR SMALL QUANTITIES, MANUFACTURED ARTICLES, RADIOACTIVE DEVICES, AND LOW SPECIFIC ACTIVITY MATERIAL

Transport Group	Small or Exempt Quantities	Manufactured Articles and Radioactive Devices Maximum Quantities		Low Specific Activity Materials
		Per Device	Per Package	
I	0.01 mCi	0.0001 Ci	0.001 Ci	0.0001 mCi/g
II	0.1 mCi	0.001 Ci	0.05 Ci	0.005 mCi/g
III	1 mCi	0.01 Ci	3 Ci	0.03 mCi/g
IV	1 mCi	0.05 Ci	3 Ci	0.3 mCi/g
V	1 mCi	1 Ci	1 Ci	
VI	1 mCi	1 Ci	1 Ci	
VII	25 Ci	25 Ci	200 Ci	
Special Form	1 mCi	0.05 Ci	20 Ci	

Table 19-6-8 also lists activity limits for manufactured articles and radioactive devices such as clocks, instruments, and electronic devices that are exempt from packaging requirements, provided they meet criteria in §173.391 (b).

Low Specific Activity Materials are uranium or thorium ores, tritium oxide, and other materials in which the activity is more-or-less uniformly distributed and in which the average estimated concentration per gram does not exceed the specification of §173.389 (c).

TRANSPORT INDEX

The Transport Index (TI) is used to compute the number of packages that may be transported in one shipment. This number is placed on the package label to designate the degree of control to be exercised by the carrier during transportation.

The TI is a carefully-computed safety indicator – a measuring unit that takes into account several different factors. For instance, a TI of 1 is assigned to a package that reads 1 mrem at 3 feet from its surface; and the TI for any other package varies in proportion to the dose at that standard distance. Thus, a dose of 10 mrem/h at that point raises the TI for the package to 10; a dose of half a millirem per hour indicates that the TI is 0.5. These indexes are the radiation TI of a package.

Because some nuclear materials are capable of sustaining a nuclear chain reaction, some additional form of control is desirable to ensure that a number of packages containing these fissile isotopes will not be assembled in one place in such a manner as to produce a critical mass and geometry. A fissionability TI may be assigned on the basis of amount, nature, and purity of the fissile material involved – making allowances if neutron-absorbing material has been incorporated to discourage chain reactions.

The assignment of a TI is fairly complicated, but the important net result is this: If the total TI for any vehicle or storage area is 50 or less, there will be an adequate safety margin to prevent any chain reaction.

The radiation TI and the fissionability TI rarely will be identical; and when they differ, the package in question is assigned the larger of the two numbers for shipping purposes. This procedure ensures safety.

The TI of each package must be noted on its warning label. Each package must bear at least two of these labels on its opposite sides.

The upper limits of radiation for a shipment depend upon the conditions under which the material is transported. Ordinarily, the radiation emitted from any package is limited by DOT regulations to no more than 200 mrem/h at 3 feet from the package surface. The normal limit is 10 mrem/h, which equals a TI of 10.

A TI of only 10 would be very difficult to achieve with a spent-fuel cask; therefore, such a shipment is made on a "sole-use" basis. When the spent-fuel cask is shipped within a closed truck or rail car, the main body of the vehicle itself becomes the "package." In this case, the radiation level of the vehicle must not exceed 200 mrem/h at its surface.

STOWAGE CONTROL

TI also is used to establish the distances packages of RAM must maintain from photographic film, people, and animals (Table 19-6-9). In this way, TI helps to provide a form of stowage control and to ensure safety.

TABLE 19-6-9

STOWAGE CONTROL FOR RAIL SHIPMENTS

Total Transport Index	Minimum Separation Distance in Feet to Nearest Undeveloped Film	Minimum Distance in Feet to Area of Persons, or Minimum Distance in Feet from Dividing Partition of a Combination Car
None	0	0
0.1 to 10.0	15	3
10.1 to 20.0	22	4
20.1 to 30.0	29	5
30.1 to 40.0	33	6
40.1 to 50.0	36	7

Table 19-6-9 establishes stowage control for rail shipments (§174.700). Tables for other modes of transportation are found in §175.00, §176.700, and §177.842 (b).

The TI system, together with tables of separation distances, provides control by the carrier over radiation exposure of package handlers and casually-exposed persons.

WARNING LABELS

Each package of radioactive material, unless exempt, must be labeled (§172.403 [f]) on two opposite sides with a distinctive warning label. Each of the three label formats bears the unique trefoil symbol, as described in §172.436

through 172.440 and as illustrated in Figure 19-6-5.

Figure 19-6-5. Package labels (a), (b), and (c).

This label alerts persons who handle packages to the precaution that the package may require special handling. If the background color of the label is all white (a "white label package"), the radiation is minimal, and nothing special is required for that package. If, however, the background of the upper half of the label is yellow (a "yellow label package"), a radiation level that requires consideration may exist at the outside of the package, and an indication of what controls must be exercised for that package is related to the Transport Index concept. If the package bears a yellow label with three stripes, the rail or highway vehicle in which it is carried must be placarded. (A placard is simply a sign placed on a vehicle.) The criteria that the shipper must consider in choosing the appropriate level are listed in Table 19-6-10. Specific labeling requirements for RAM are given in §172.403.

TABLE 19-6-10

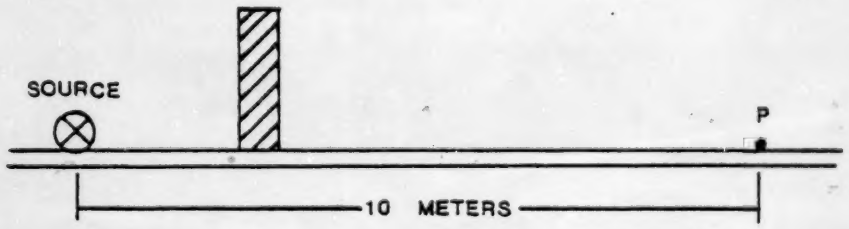
RADIOACTIVE MATERIAL PACKAGES LABELING CRITERIA
DOSE RATE LIMITS

Label	At Any Point on Accessible Surface of Package	At Three Feet From External Surface of Package (Transport Index)
"Radioactive-White I"	0.5 mR/h	0
"Radioactive-Yellow II"	50 mR/h	1.0 mR/h
"Radioactive-Yellow III"*	200 mR/h	10 mR/h

*Requires Vehicle Placarding
(This label is mandatory for any Fissile Class III (173.389A) or large quantity package (173.389B), regardless of dose rate levels.)

EXERCISES

1. Find the exposure rate caused by skyshine at P for a 150 Ci ^{60}Co source.



2. Determine the fraction of incident beta-particle energy converted to bremsstrahlung for a ^{32}P source and a copper absorber.
3. Calculate the maximum total dose received by a person standing six feet from the following radioactive mate-

rial packages having the TI values indicated.

a. Package 1: TI = 10.
b. Package 2: TI = 1.
c. Package 3: TI = 6.

MATERIALS

Portable GM survey meter.
Scintillation probe with scalar.
^{90}Sr and ^{32}P sources.
Shielding material: lead, copper, and aluminum sheets.

PROCEDURES

1. Place the ^{90}Sr source on a tabletop, and move the GM survey meter towards it until an approximate half-scale reading on the low range is obtained. Record the distance and the count rate in Data Table 19-6-1.
2. Record the count rate with various shielding materials placed between the source and the survey meter.
3. Repeat Step 2 with the scintillation detector, and record instrument response.
4. Using the ^{90}Sr source and scintillation detector, place a combination shield of lead and aluminum between the source and the detector. Record the count rate; and then reverse the positions of the lead and aluminum, and again record the count rate.

5. Repeat Steps 1 through 4 with the ^{32}P source.

Questions:

1. Explain the difference in GM and scintillation response when different shields are used.
2. Why is the scintillation response not constant when the lead and aluminum are changed in position (Step 4)?
3. Compare the bremsstrahlung measurements for different Z materials. Does instrument response correspond with the following equation?

$$f = 3.5 \times 10^{-4} Z E$$

4. Compare the amount of bremsstrahlung produced by ^{90}Sr with that produced by the ^{32}P betas. Does bremsstrahlung production correspond to the above equation?

DATA TABLE 19-6-1

1. (Step 1)

Detector	Instrument Response	
	^{90}Sr	^{32}P
GM		
Scint.		

2. (Steps 2 and 3)

Source	Detector	Shielding Material	Z	Inst. Response
^{90}Sr	GM	Lead		
		Copper		
		Aluminum		
^{90}Sr	Scint.	Lead		
		Copper		
		Aluminum		
^{32}P	GM	Lead		
		Copper		
		Aluminum		
	Scint.	Lead		
		Copper		
		Aluminum		

3. (Step 4)

Shields	Inst. Response
Lead-aluminum	
Aluminum-lead	

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SECTION I: Matching.

1. Match the term in column A with the correct definition in column B.

"A"		"B"	
____	1. Normal Form Material.	A.	Indicates that a radiation level that requires consideration exists.
____	2. Special Form Material.	B.	Indicates that radiation is minimal.
____	3. Transport group.	C.	Used to compute the number of packages that may be shipped in one shipment.
____	4. Type "A" package.	D.	Materials capable of sustaining a nuclear chain reaction.
____	5. Type "B" package.	E.	Materials in which activity is uniformly distributed, more-or-less.
____	6. Small Quantity Material.	F.	Material exempt from general packaging requirements.
____	7. Low Specific Activity Material.	G.	Does not need specific prior approval by DOT.
____	8. Transport Index.	H.	Able to withstand a 30-foot free drop onto an unyielding surface.
____	9. White label package.		
____	10. Yellow label package.		

I. Present little hazard due to radiotoxicity.
J. May be encapsuled as a sealed source.
K. More than 1,000 Ci of a transport group VI material.

SECTION II.

Select the best response(s) to the following questions by circling the alphabetical designation:

2. What is the best combination of materials for the shielding of a mixed neutron and gamma emitter?
   a. Aluminum and lead.
   b. Lead and plastic.
   c. Paraffin and aluminum.
   d. Water and lead.
3. Which shielding is best for situations in which bremsstrahlung is stronger than gamma flux?
   a. Lead and concrete.
   b. Aluminum and lead.
   c. Water and lead.
   d. Aluminum and paraffin.
4. When lead shielding is constructed, joints of lead sheets –
   a. Should overlap one meter.
   b. Must be fastened to masonry columns.
   c. Should overlap at least 1 cm.
   d. Should leave space for the installation of ducting.

5. When a beta particle collides inelastically with a nucleus, the degree of hazard is a function of which of the following?
   a. The specific gravity of the material with which it collides.
   b. The fraction of incident beta radiation converted into photons.
   c. The atomic number of the absorber.
   d. The maximum energy of the beta.
6. Neutron radiation –
   a. Cannot produce secondary gamma rays.
   b. Produces X radiation only.
   c. May produce secondary gamma rays by neutron capture.
   d. May give rise to mixed radiations.

SECTION III: Essay.

7. List three ways in which mixed radiations occur naturally.
   a.

   b.

   c.

8. Discuss the structural aspects of lead and masonry shielding.

9. Specify types of shielding for –
   a. A mixed alpha and gamma field.

   b. A mixed neutron and gamma field.

   c. A gamma and X-ray field.

10. Describe how radioactive material is classified according to form, transport group, and quantity.

11. Describe how the Transport Index is assigned to a package of radioactive material.

12. List the types of warning labels, and describe how they are placed on packages of radioactive material.

13. How is the Transport Index used to establish stowage control?

SECTION IV: Calculations.

14. Write an equation that expresses the relationship between the fraction of the incident beta radiation converted into photons, the atomic number of the absorber, and the maximum energy of the incident beta particles.

15. Given: The 20-Ci source of ^{60}Co at a distance of 20 meters from a point.
Find: The ER due to skyshine at P.
The gamma-ray constant of ^{60}Co is 1.32 R/h $\frac{m^2}{C}$.

NUCLEAR TECHNOLOGY

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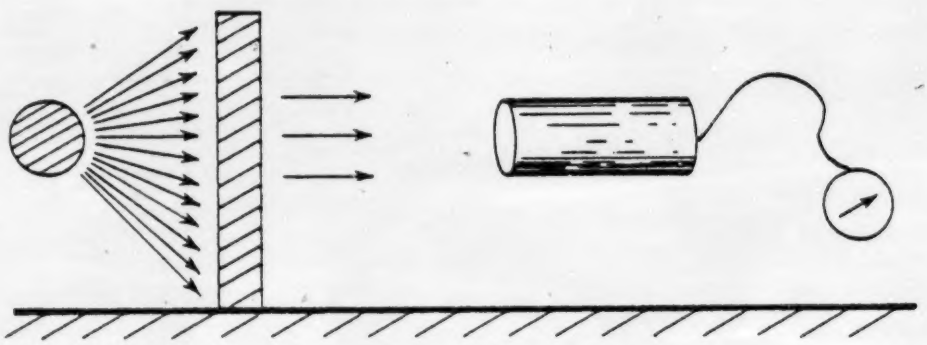
COURSE 19 - RADIATION SHIELDING



RADIATION SHIELDING
COURSE 19

ORD CENTER FOR OCCUPATIONAL RESEARCH AND DEVELOPMENT

NUCLEAR TECHNOLOGY SERIES

INDEX

for

COURSE 19, RADIATION SHIELDING (SIX MODULES)

NOTE: Each module in this Course is numbered separately, and the page numbers begin with page 1 in each module. Therefore, this Index lists for each entry the module number (underlined), followed by a comma and the page number. Figures and tables are listed by their numbers, followed by the module and page number in parentheses. A Master Index is available in a separate volume. This general index merges all index entries from the 35 Nuclear Courses. It is a valuable research tool, with which the student or researcher can locate all the references to any subject within the 35 Courses.

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